CHAPTER 5
CONTAINMENT SYSTEMS

5.1 CONTAINMENT STRUCTURES

5.1.1 Design Basis

The reactor containment completely encloses the entire reactor and reactor coolant system and ensures that essentially no leakage of radioactive materials to the environment would result even if gross failure of the reactor coolant system were to occur. The liner and penetrations are designed to prevent any leakage through the containment. The structure provides biological shielding for both normal and accident situations.

The reactor containment is designed to safely withstand several conditions of loading and their credible combinations. The major loading conditions are:

1. Occurrence of a gross failure of the reactor coolant system, which creates a high-pressure and temperature condition within the containment.

2. Coincident failure of the reactor coolant system with an earthquake or wind.

5.1.1.1 Principal Design Criteria

5.1.1.1.1 Quality Standards

Criterion: Those systems and components of reactor facilities, which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents, which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required. (GDC 1)

The containment system structure is of primary importance with respect to its safety function in protecting the health and safety of the public.

Quality standards of material selection, design, fabrication, and inspection governing the above features conforms to the applicable provisions of recognized codes and good nuclear practice. The concrete structure of the reactor containment conforms to the applicable portions of ACI−18-63. Further elaboration on quality standards of the reactor containment is given is Section 5.1.1.5.

5.1.1.1.2 Performance Standards

Criterion: Those systems and components of reactor facilities, which are essential to the prevention or to the mitigation of the consequences of nuclear accidents, which cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that enable such systems and components to withstand, without undue risk to the health and safety of the public, the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design. (GDC 2)

All components and supporting structures of the reactor containment are designed so that there is no loss of function of such equipment in the event of maximum potential ground acceleration acting in the horizontal and vertical directions simultaneously. The dynamic response of the structure to ground acceleration, based on the site characteristics and on the structural damping, is included in the design analysis. The reactor containment is defined as a Class I structure for purposes of seismic design (Section 1.11). Its structural members have sufficient capacity to accept, without exceeding specified stress limits, a combination of normal operating loads, functional loads due to a loss-of-coolant accident, and the loadings imposed by the maximum potential earthquake.

5.1.1.1.3 Fire Protection

Criterion: A reactor facility shall be designed to ensure that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used throughout the facility wherever necessary to preclude such risk, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.(GDC 3)

Fire protection in all areas of the nuclear electric plant is provided by structure and component design that optimizes the containment of combustible materials and maintains exposed combustible material below the ignition temperature. The station is designed on the basis of limiting the use of combustible materials in construction by using fire-resistant materials to the greatest extent practical. Containment liner thermal insulation does not support combustion. The bearing oil systems for the reactor coolant pump motors are self-contained.

All oil-containing equipment associated with the reactor coolant pump motors is also completely enclosed by an oil-collecting system, which in the event of an oil leak, will contain and channel away the oil to remote storage containers.

5.1.1.1.4 Records Requirement

Criterion: The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public. (GDC 5)

Records of the design, fabrication, construction, and testing of the reactor containment are maintained throughout the life of the reactor.

5.1.1.1.5 Reactor Containment

Criterion: The containment structure shall be designed (a) to sustain, without undue risk to the health and safety of the public, the initial effects of gross equipment failures, such as a large reactor coolant pipe break, without loss of required integrity, and (b) together with other engineered safety features as may be necessary, to retain for as long as the situation requires, the functional capability of the containment to the extent necessary to avoid undue risk to the health and safety of the public. (GDC 10).

The design pressure and temperature of the containment exceeds the peak pressure and temperature occurring as the result of the complete blowdown of the reactor coolant through any rupture of the reactor coolant system up to and including the hypothetical double-ended severance of a reactor coolant pipe. Energy contribution from the steam system is included in the calculation of the containment pressure transient due to reverse heat transfer through the steam generator tubes. The supports for the reactor coolant system are designed to withstand the blowdown forces associated with the sudden severance of the reactor coolant piping so that the coincidental rupture of the steam system is not considered credible.

In 1989, the NRC approved changes to the design bases with respect to dynamic affects of postulated primary loop pipe ruptures, as discussed in Section 4.1.2.4.

The containment structure and all penetrations are designed to withstand, within design limits, the combined loadings of the design-basis accident and design and maximum potential seismic conditions.

All piping systems that penetrate the vapor barrier are anchored at the liner. The penetrations for the blowdown, and sample lines are designed so that the penetration is stronger than the piping system and so that the vapor barrier is not breached due to a hypothesized pipe rupture. The pipe rupture loads for the main steam and feedwater lines are resisted by the supports located away from their penetrations and do not affect the integrity of the penetrations for these lines. The pipe capacity in flexure is assumed to be limited to the plastic moment, based upon the yield strength of the pipe material. All lines (with the exception of sample tubing) connected to the primary coolant system that penetrate the vapor barrier are also restrained near the secondary shield walls (i.e., walls surrounding the steam generators and reactor coolant pumps) and are each provided with at least one valve between the shield wall and the reactor coolant system. These restraints are designed to withstand the thrust, moment, and torque resulting from a hypothesized rupture of the attached pipe.

All isolation valves are supported to withstand, without impairment of valve operability, the combined loadings of the design basis accident and design and maximum potential seismic conditions.

Section 5.1.5 includes a discussion of the details of the design of primary system supports. In addition, the design pressure will not be exceeded during any subsequent long-term pressure transient determined by the combined effects of heat sources, such as residual heat and limited metal-water reactions, structural heat sinks, and the operation of the engineered safeguards, which uses only the emergency electric power supply.

5.1.1.1.6 Reactor Containment Design Basis

Criterion: The reactor containment structure, including openings and penetrations, and any necessary containment heat removal systems, shall be designed so that the leakage of radioactive materials from the containment structure under conditions of pressure and temperature resulting from the largest credible energy release following a loss-of-coolant accident, including the calculated energy from metal-water or other chemical reactions that could occur as a consequence of failure of any single active component in the emergency core cooling system, will not result in undue risk to the health and safety of the public. (GDC 49)

The following general criteria are followed to ensure conservatism in computing the required structural load capacity:

1. In calculating the containment pressure, rupture sizes up to and including a double-ended severance of reactor coolant pipe are considered.

2. In considering postaccident pressure effects, various malfunctions of the emergency systems are evaluated. Contingent mechanical or electrical failures are assumed to disable one of the diesel generators, two of the five fan-cooler units, and one of the two containment spray units. Equipment, which can be run from diesel power is described in Chapter 8.

3. The pressure and temperature loadings obtained by analyzing various loss-of-coolant accidents, when combined with operating loads and maximum wind or seismic forces, do not exceed the load-carrying capacity of the structure, its access opening or penetrations.

The most stringent case of these analyses is summarized in Section 14.3.5.3.7.

5.1.1.1.7 Nil-ductility Transition Temperature Requirement for Containment Material

Criterion: The selection and use of containment materials shall be in accordance with applicable engineering codes. (GDC 50).

The selection and use of containment materials comply with the applicable codes and standards tabulated in Section 5.1.1.5.

The concrete containment is not susceptible to a low-temperature brittle fracture.

The containment liner is enclosed within the containment and thus is not exposed to the temperature extremes of the environs. The containment ambient temperature during operation is between 50 and 130°F. This includes both hot operating and cold shutdown conditions. The Containment liner was specified for impact testing at a temperature of 30°F below the minimum service temperature of 50°F. The large Containment steel penetrations, equipment hatch and personnel lock, were specified for impact testing at -50°F which is more than 30°F below the outside containment temperature of -5°F. These tests assure that the Nil Ductility Transition Criterion of GDC 50 is met.

5.1.1.2 Supplementary Accident Criteria

Systems relied upon to operate under postaccident conditions, which are located external to the containment and communicate directly with the containment, are considered to be extensions of the leakage boundary.

The pressure retaining components of the containment structure are designed for the maximum potential earthquake ground motion of the site combined with the simultaneous loads of the design basis accident as follows:

1. The liner is designed to ensure that no average strains greater than the strain at the guaranteed yield point occur at the factored loads. In regions of local stress concentrations or stresses due to localized secondary load effects, the liner is permitted to yield but the maximum liner strain is limited to 0.5-percent.

2. The mild steel reinforcement is designed to ensure that no strains greater than the strain at the guaranteed yield point occur at a cross section under the factored loads. The local yielding of reinforcing bars are permitted around the large openings for load combinations that include seismic loads.

The pressure-retaining components of containment subject to deterioration or corrosion in service are provided with appropriate protective means or devices (e.g., protective coatings).

5.1.1.3 Energy and Material Release

The design pressure is not exceeded during any subsequent long-term pressure transient determined by the combined effects of heat sources such as residual heat and metal-water reactions, structural heat sinks, and the operation of other engineered safety features utilizing only the emergency onsite electric power supply. The mass and energy releases to and the accident pressure and temperature effects on the containment structures, are those created by the hypothetical large break loss-of-coolant accident as presented in Section 14.3.5.

The following loadings are considered in the design of the containment in addition to the pressure and temperature conditions described above:

1. Structure dead load.
2. Live loads.
3. Equipment loads.
4. Internal test pressure.
5. Earthquake.
6. Wind.

5.1.1.4 Engineered Safety Features Contribution

Five types of engineered safety features are included in the design of this facility to ensure containment integrity. These systems are discussed in Chapter 6 and their effectiveness is analyzed in Chapter 14.

5.1.1.5 Codes and Standards

The design, materials, fabrication, inspection, and proof testing of the containment vessel complies with the applicable parts of the following codes and standards.

Code		Title
1.	ASTM A-333, Gr. 1	Specification for Seamless and Welded Steel Pipe for Low Temperature Service
2.	ASTM A-181	Forged or Rolled Steel Pipe Flanges, Forged Fittings, and Valves and Parts for General Service
3.	ASTM A-300, Cl. 1, Firebox	Specification for Notch Toughness Requirements for Normalized Steel Plates for Pressure Vessels
4.	ASTM A-201, Gr. B	Specification for Carbon Silicon Steel Plates of Intermediate Tensile Ranges for Fusion Welded Boilers and other Pressure Vessels
5.	ASTM A-36	Specification for Structural Steel
6.	ASTM A-131, Gr. C	Specification for Structural Steel for Ships
7.	ASTM A-240	Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Fusion-Welded Unfired Pressure Vessels
8.	ASTM A-312	Specification for Seamless and Welded Austenitic Stainless Steel Pipe
9.	ASTM A442, Grade 60	Specifications for Pressure Vessel Plates, Carbon Steel, Improved Transition Properties
10.	ASME Boiler and Pressure Nuclear Vessels Vessel Code-Section III	Nuclear Vessels
11.	ASME Boiler and Pressure Unfired Pressure Vessels Vessel Code-Section VIII	Unfired Pressure Vessels
12.	ASME Boiler and Pressure Welding Qualifications Vessel Code-Section IX	Welding Qualifications
13.	ASTM C-33	Standard Specifications for Concrete Aggregates
14.	ASTM C-150	Standard Specifications for Portland Cement
15.	ASTM C-172	Standard Method of Sampling Fresh Concrete
16.	ASTM C-31	Standard Method of Making and Curing Concrete Compression and Flexure Test Specimens in the Field
17.	ASTM C-39	Standard Method of Test for Compressive Strength of Molded Concrete Cylinders
18.	ASTM-C-350	Specifications For Fly Ash For Use As AN Admixture in Portland Cement Concrete
19.	ASTM C-94	Specifications for Ready Mixed Concrete

20.	ASTM C-42	Standard Methods of Securing, Preparing, and Testing Specimens from Hardened Concrete for Compressive and Flexural Strengths
21.	ASTM C-494	Specifications for Chemical Admixtures for Concrete
22.	ASTM A-305	Specifications for Minimum Requirements for Deformations of Deformed Steel Bars for Concrete Reinforcement
23.	ASTM A-408	Specifications for Special Large Size Deformed Billet-Steel Bars for Concrete Reinforcement
24.	ASTM A-432	Specification for Deformed Billet Steel Bars for Concrete Reinforcement with 60,000 psi Minimum Yield Strength
25.	Research Council of Riveted and Bolted Structural Joints of the Engineering Foundation	Specification For Structural Joints Using ASTM A-325 Bolts
26.	ACI-613	Recommended Practice for Selecting Proportions for Concrete
27.	ACI-306	Recommended Practice for Winter Concreting
28.	ACI-318, Part IV-B	Structural Analysis and Proportioning of Members-Ultimate Strength Design
29.	ACI-318	Building Code Requirements for Reinforced Concrete
30.	ACI- 505	Specification for the Design and Construction of Reinforced Concrete Chimneys
31.	ACI-315	Manual of Standard Practice for Detailing Reinforced Concrete Structures
32.	ASA N6.2	Safety Standards for the Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors
33.	ASA A58.1	American Standard Code Requirements for Minimum Design Loads in Buildings and Other Structures
34.		State Building and Construction Code for the State of New York
35.	SSPC-SP-6	Commercial Blast Cleaning

5.1.2 Containment Structure Design

5.1.2.1 General Description

The reactor containment structure is a reinforced concrete vertical right cylinder with a flat base and hemispherical dome. A welded steel liner with a minimum thickness of 0.25-in. is attached to the inside face of the concrete shell to ensure a high degree of leaktightness. The design objective of the containment structure is to contain all radioactive material, which might be released from the core following a loss-of-coolant accident. The structure serves as both a biological shield and a pressure container.

The structure, as shown on Plant Drawings 9321-2501, 9321-2502, 9321-2503, 9321-2506, 9321-2507, 9321-2508, [Formerly UFSAR Figures 5.1-2 through 5.1-7] and Figures 5.1-1 consists of side walls measuring 148-ft from the liner on the base to the springline of the dome, and has an inside diameter of 135-ft. The side walls for the cylinder and the dome are 4-ft 6-in. and 3-ft 6-in. thick respectively. The inside radius of the dome is equal to the inside radius of the cylinder so that the discontinuity at the springline due to the change in thickness is on the outer surface. The cylindrical part of the liner is substantially round. The difference between the minimum and maximum inside diameters at any selected cross section does not generally exceed 0.25-percent of the nominal diameter at that elevation. Between elevations 43-ft and 95-ft, the maximum diameter of any cross section is 135-ft 2-in., and the minimum diameter is 134-ft 10-in. except at the liner closing the temporary opening in the northwest quadrant where a minimum diameter of 134-ft 8-5/8-in. was measured. This portion of the liner was erected after all exterior concrete work was completed and is within the local buckle allowance of the liner plates. Above elevation 95 ft the tolerance on inside diameter does not exceed 0.50-percent of the nominal diameter of the selected cross section. The liner is erected true and plumb so that the deviation does not exceed 1/500 of the height at the selected cross section (allowing for 2-in. local buckling of the liner plates).

Particular care is taken in matching edges of cylindrical and hemispherical sections to ensure that all joints are properly aligned. Maximum permissible offset of completed joints is 25 percent of nominal plate thickness. Plates buckled beyond acceptable limits are cut out and replaced with new plates.

The flat concrete base mat is 9-ft thick with the bottom liner plate located on top of this mat. The bottom liner plate is covered with 3-ft of concrete, the top of which forms the floor of the containment.

Where uplift from pressure occurs at the outer areas of the mat, the 9-ft thick mat has sufficient flexural capacity to resist the uplift.

No hydraulic uplift exists since the bottom elevation of the mat is considerably higher than that of the high water level.

The large mass of the containment including interior concrete and equipment makes the structure inherently stable from overturning due to seismic motion.

In addition, keying action from the reactor pit and sumps, plus friction between the concrete and rock, prevents a sliding of the structure from horizontal ground motion.

The basic structural elements considered in the design of the containment structure are the base slab, side walls, and dome acting as one structure under all possible loading conditions. The liner is anchored to the concrete shell by means of stud anchors. The lower portions of the cylindrical liner are insulated to avoid thermal deformation of the liner under accident conditions.

The containment structure is inherently safe with regard to common hazards such as fire, flood, and electrical storm. The thick concrete walls are invulnerable to fire and only an insignificant amount of combustible material, such as lubricating oil in pump and motor bearings, is present in the containment.

Internal structures consist of equipment supports, shielding, reactor cavity and canal for fuel transfer, and miscellaneous concrete and steel for floors and stairs. All internal structures are supported on the mat with the exception of equipment supports secured to the intermediate floors.

A 3-ft thick concrete ring wall serving as a missile and partial radiation shield surrounds the reactor coolant system components and supports the polar-type reactor containment crane. A 2-ft thick reinforced concrete floor covers the reactor coolant system with removable gratings in the floor provided for crane access to the reactor coolant pumps. The four steam generators, pressurizer, and various piping penetrate the floor. Spiral stairs provide access to the areas below the floor.

The refueling canal connects the reactor cavity with the fuel transport tube to the spent fuel pool. The floor and walls of the canal are concrete, with wall and shielding water providing the equivalent of 6-ft of concrete.

The refueling canal floor is 5-ft thick. The concrete walls and floor are lined with 0.25-in. thick stainless steel plate. The linings provide a leakproof membrane that is resistant to abrasion and damage during fuel handling operation.

Waterproofing is provided in the areas of the containment in contact with backfill to prevent ground-water seepage. This consists of a coat of bitumastic No. 50, a 0.625-in.-thick layer of hardboard insulation, and a second coat of bitumastic No. 50. Fill for innermost 5-ft from containment walls is crushed rock of maximum size of 6-in. and minimum amount of fines. All fill is free of vegetable matter.

5.1.2.2 Design Load Criteria

The following loads are considered to act upon the containment structure creating stresses within the component parts.

1. Dead load consists of the weight of the concrete wall, dome, liner, insulation, base slab, and the internal concrete. Weights used for dead load calculations are as follows:

    a. Concrete — 150 lb/ft^3

    b. Reinforcing steel — 490 lb/ft^3 using nominal cross-sectional areas of reinforcing as defined in ASTM for bar sizes.

    c. Steel lining — 490 lb/ft^3 using nominal cross-sectional area.

    d. Insulation — 6 lb/ft^3 including stainless steel jacket.

2. Live load consists of snow and construction loads on the dome and major components of equipment in the containment. Snow and ice loads are assumed to be applied uniformly to the top surface of the dome at an estimated value of 20 lb/ft^2 of horizontal projection of the dome. This loading represents approximately 2-ft of snow, which is considered to be a conservative amount since the slope of the dome will tend to cause much of the snow that falls on it to slide off. A

construction live load of 50 lb/ft^2 has been used on the dome, but will not be considered to act concurrently with the snow load. Equipment loads are considered as specified on the drawings supplied by the manufacturers of the various pieces of equipment.

Design live loads inside the containment building are as follows:

a. Elevation 68-ft-0-in. 10-ft strip adjacent to crane wall = 600 psf Remaining strip = 100 psf

b. Elevation 95-ft-0-in. Concrete slab = 500 psf Grating areas = 100 psf

3. The internal pressure transient used for the containment design and its variation with time is based on a postulated large break LOCA of 47 psig and liner temperature of 247°F. For the free volume of 2,610,000-ft^3 within the containment, the design pressure is 47 psig. This pressure transient is more severe than those calculated for various loss-of-coolant accidents, which are presented in Section 14.3.

4. Thermal expansion stresses due to an internal temperature increase caused by a loss-of-coolant accident is considered. The maximum temperature at the uninsulated section of the liner under accident conditions is 247°F. For the 1.25 times and 1.50 times design pressure loading conditions given in Section 5.1.2.4, the corresponding liner temperatures will be 285°F and 306°F respectively. The minimum external ambient design temperature, averaged over a 24 hour period, is 0°F. The liner maximum temperature following a loss-of-coolant accident with an outside air temperature of 0°F was calculated to be less than 247°F at the Stretch Power Uprate (SPU) power rating of 3216 MWt for the core. The initial containment air temperature in the SPU analysis for the liner temperature was set to 110°F, which is the maximum expected operating temperature at 100% power with an outside temperature of 0°F.

5. The ground acceleration for the design earthquake has been determined to be 0.1g applied horizontally and 0.05g applied vertically. These values have been resolved as conservative numbers based upon recommendations from Dr. Lynch, Director of Seismic Observatory, Fordham University.

   A dynamic analysis is used to arrive at equivalent design loads. Additionally, a hypothetical ground acceleration of 0.15 g horizontal and 0.10 g vertical is used to analyze for the no-loss-of-function. This is discussed in Section 5.1.3.11, Seismic Design.

   Due to symmetry of the containment structure, torsional loads generated by an earthquake are insignificant and have not been considered.

   Tornado loads have not been considered in the design of the Unit 2 containment; however, the seismic bars provide a more than adequate mechanism to withstand the torsional effect if it were to occur. An evaluation of the effect of tornado loads on the containment structure is presented in Appendix B of the Containment Design Report.

6. The American Standards Association "American Standard Code Requirements for Minimum Design Loads in Buildings and Other Structures" (A58.1-1955) designates the site as being in a 25 psf zone for wind loads. In this code, for height zones between 100 and 499-ft, the recommended wind pressure on a flat surface is 40 psf. Correcting for the shape of the containment by using a shape factor of 0.60, the recommended pressure becomes 24 psf. The state building and construction code for the State of New York stipulates a wind pressure up to 30 psf on a flat surface for heights up to 300 feet. For design, a 30 psf basic wind load has been used from ground level up.

7. Internal pressure was applied to test the structural integrity of the containment shell up to 115-percent of the design pressure. For this structure, the test pressure is 54 psig. The containment is also structurally designed to withstand an external pressure 2.5 psig higher than the internal pressure.

5.1.2.3 Material Specifications

Basically five materials are used for the construction of the containment structure. These are:

1. Concrete.
2. Reinforcing steel.
3. Plate steel liner.
4. Insulation.
5. Protective Coating.

Basic specifications for these materials are as follows:

1. Concrete is a dense, durable mixture of sound coarse aggregate, fine aggregate, cement, and water. Cement conforms to ASTM, Specification C-150-65 "Standard Specification for Portland Cement," Type I (Normal), or Type II (moderate heat of hydration) requirements. Whenever high early strength is required, Type III Cement is used. Water is free from any injurious amounts of acid, alkali, salts, oil, sediment, or organic matter. The concrete has a minimum density of 150 lb/ft^3. The 28-day standard compressive strength of the concrete is 3000 psi. Adequate means of control are used in the manufacture of the concrete. To ensure the values of compressive strength are attained as a minimum, concrete samples are tested in accordance with the following ASTM Standards:

   ASTM C-172 - Standard Method of Sampling Fresh Concrete

   ASTM C-31 - Standard Method of Making and Curing Concrete Compression and Flexure Test Specimens in Field

   ASTM C-39 - Standard Method of Test for Compressive Strength of Molded Concrete Cylinders

All making and testing of concrete samples have been performed by Vacca Testing Laboratory and Research Company, Inc.

At certain specifically evaluated locations, non-structural surface type cracks and delaminations in the containment concrete have been repaired by injection of engineering approved epoxy grout. Although non-structural in nature, these repairs were performed in accordance with the requirements of IWL-4210 of the 1992 ASME Boiler and Pressure Vessel Code, Section XI, as applicable.

2. Reinforcing steel for the dome, cylindrical walls and base mat is high-strength, deformed billet steel bars conforming to ASTM Designation A432-65 "Specification for Deformed Billet Steel Bars for Concrete Reinforcement with 60,000 psi Minimum Yield Strength." This steel has a minimum yield strength of 60,000 psi, a minimum tensile strength of 90,000 psi, and a minimum elongation of 7-percent in an 8-in. specimen. Reinforcing bars No. 11 and smaller in diameter are lapped spliced in the mat for flexural loadings and spliced by the Cadweld process in the walls and dome for tension loading. Bars No. 14S and 18S are spliced by the Cadweld process only. A certification of physical properties and chemical content of each heat of reinforcing steel delivered to the job site has been issued from the steel supplier. The splices used to join reinforcing bars have been tested to ensure that they will develop at least 125-percent of the minimum yield point stress of the bar. The test program required cutting out, at random, approximately 3-percent, completed splices and testing to determine their breaking strength.

3. The plate steel liner is carbon steel conforming to ASTM Designation A442-65 "Standard Specification for Carbon Steel Plates with Improved Transition Properties," Grade 60. This steel has a minimum yield strength of 32,000 psi and a minimum tensile strength of 60,000 psi with an elongation of 22-percent in an 8-in. gauge length at failure.

   The liner is 0.25-in. thick at the bottom, 0.50-in. thick in the first three courses, except 0.75-in. thick at penetrations, a minimum of 0.34-in. in the general area at elevation 46-ft. due to past corrosion, and 0.375-in. thick for remaining portion of the cylindrical walls and 0.50-in. thick in the dome. The 0.34-in. minimum thickness affects the calculated stress levels presented in the Containment Design Report and the Containment Liner Stress Analysis Report. However, evaluation of the reduced minimum thickness has concluded that no design criteria are exceeded. The liner material has been tested to ensure an NDTT more than 30°F lower than the minimum operating temperature of the liner material.

   Impact testing has been done in accordance with Section N331 of Section III of the ASME Boiler and Pressure Vessel Code. A 100-percent visual inspection of liner anchors was made prior to pouring concrete.

4. The material used for the original insulation of the liner plate was polyvinylchloride with stainless steel jacket. This insulation has been selected to withstand the calculated temperature and pressure conditions associated with a postulated large break LOCA of 47 psig and liner temperature of 247°F. The carbon steel liner with an inorganic zinc protective coating makes contact with

the polyvinylchloride insulation, the stainless steel, and the sealant. However, these materials do not react with each other.

Because the insulation panels are jacketed with stainless steel and sealed at the joints, the insulation will not be subjected to the moisture and high humidity atmosphere of the containment during an accident.

Manufacturer's tests on the polyvinylchloride insulation indicated that the insulation was capable of withstanding periodic compression at 60 psig at temperatures from 40°F to 120°F and a single compression under accident conditions without any detriment or change to the insulation properties. The manufacturer's analog transient analysis indicated only a 5°F rise in liner temperature 1000 sec after an exposure to 310°F for the entire duration of the analysis. This provides a factor of safety of approximately 15 on specified tolerable temperature rise in the liner. A factor of safety of 2 is provided on specified insulation performance versus tolerable temperature rise in liner.

The maximum normal operating temperature of the containment was changed from 120°F to 130°F by Amendment 149 to the Facility Operating License DPR-26 for IP-2 dated March 27,1990. Evaluations performed show the insulation material used on the containment liner is adequate for use at the higher operating temperature.

For additional information on the liner insulation and the modifications made to it in 1973, see Section 5.1.7.

5. One 3 mil shop coat of Carbozinc No. 11 primer and one 4 mil minimum finish coat of Phenoline No. 305 as manufactured by the Carboline Company have been applied to the liner, as well as essentially all painted surfaces in containment, in accordance with the manufacturer's recommendations.

   The effect of the postaccident environment on protective coatings was conservatively evaluated for Indian Point Unit 2. The coatings showed no deterioration after a number of cycles. A more thorough discussion on the qualifications of the protective coatings applied during construction is presented in WCAP-7198-L.[1]

   In addition, various areas inside containment have been repaired and recoated with other DBA qualified coatings approved for use at Indian Point 2. Protective coatings used inside the containment are procured, applied, and maintained in compliance with Regulatory Guide 1.54 (June 1973), “Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants.” New quality requirements will be developed based on it’s provisions, but specific requirements, such as documented site meetings, field demonstrations, substrate priming, applicator reporting, inspection reporting and report forms will be considered on a job-by-job basis.

Quality of both materials and construction of the containment structure was ensured by a continuous program of quality control and inspection by Con Edison, and/or its field representatives, and Westinghouse Atomic Power Division, and United Engineers and Constructors Inc., as described in Section 5.1.2.6.

5.1.2.4 Design Stress Criteria

The design is based upon limiting load factors that are used as the ratio by which loads will be multiplied for design purposes to ensure that the loading deformation behavior of the structure is one of elastic, tolerable strain behavior. The load factor approach is being used in this design as a means of making a rational evaluation of the isolated factors, which must be considered in ensuring an adequate safety margin for the structure. This approach permits the designer to place the greatest conservatism on those loads most subject to variation and which most directly control the overall safety of the structure. In the case of the containment structure, therefore, this approach places minimum emphasis on the fixed gravity loads and maximum emphasis on accident and earthquake or wind loads. The loads utilized to determine the required limiting capacity of any structural element on the containment structure are computed as follows:

1. $C = 1.0D \pm 0.05D + 1.5\,P + 1.0\,(T + TL)$
2. $C = 1.0D \pm 0.05D + 1.25\,P + 1.0\,(T' + TL') + 1.25E$
3. $C = 1.0D \pm 0.05D + 1.0P + 1.0\,(T'' + TL'') + 1.0E'$

Symbols used in these formulae are defined as follows:

C = Required load capacity of section.

D = Dead load of structure and equipment loads.

P = Accident pressure load as shown on pressure-temperature transient curves.

T = Load due to maximum temperature gradient through the concrete shell and mat based upon temperature associated with 1.5 times accident pressure.

TL = Load exerted by the liner based upon temperatures associated with 1.5 times accident pressure.

T' = Load due to maximum temperature gradient through the concrete shell and mat based upon temperatures associated with 1.25 times accident pressure.

TL' = Load exerted by the liner based upon temperatures associated with 1.25 times accident pressure.

E = Load resulting from either design earthquake or wind, whichever is greater.

T" = Load due to maximum temperature gradient through the concrete shell and mat based upon temperatures associated with the accident pressure.

TL" = Load exerted by the liner based upon temperatures associated with the accident pressure.

E' = Load resulting from assumed hypothetical earthquake.

A chart for allowable versus actual stresses has been included in the Containment Design Report.

Load condition (1) indicates that the containment will have the capacity to withstand loadings at least 50-percent greater than those calculated for the postulated loss-of-coolant accident alone. Results of analysis using load condition (1) are shown in Figure 5.1-11.

Load condition (2) indicates that the containment will have the capacity to withstand loadings at least 25-percent greater than those calculated for the postulated loss-of-coolant accident with a coincident design earthquake. Results of analysis using load condition (2) are shown in Figure 5.1-12.

Load condition (3) indicates that the containment will have the capacity to withstand loads at least equal to those calculated for the postulated loss-of-coolant accident with a coincident hypothetical earthquake defined in Section 5.1.2.2. Results of analysis using load condition (3) are shown in Figure 5.1-13.

The mat has been analyzed using load conditions (1), (2) and (3) as shown in Figures 5.1-14 through 5.1-16 and also for loads occurring only at operating and test pressure conditions. For loads, see Table 5.1-1, Flooded Weights-Containment Building.

The loads resulting from wind on any portion of the structure do not exceed those resulting from earthquake.

The capacity of all structural components, with the minor exceptions of outer rebar at large containment openings addressed in Section 3.4.4 of the Containment Design Report, exceeds or is equal to the capacity required by the most severe loading combination. The loads resulting from the use of these equations will hereafter be termed "factored loads."

The load factors used in these equations are based upon the load factor concept employed in Part IV-B, "Structural Analysis and Proportioning of Members Ultimate Strength Design" of ACI –318-63. Because of the refinement of the analysis and the restrictions on construction procedure, the load factors in the design primarily provide for a safety margin on the load assumptions.

The design includes the consideration of both primary and secondary stresses. The design limit for tension member (i.e., the capacity required for the design load) is based upon the yield stress of the reinforcing steel.

The theoretical load carrying capacity of steel reinforced concrete cross-sections are reduced by a capacity reduction factor "∅", which provides for the possibility that small adverse variations in material strengths, workmanship, dimensions, and control, while individually within required tolerances and the limits of good practice, occasionally may combine to result in under-capacity. For tension members, the factor "∅" has been established as 0.95. The factor "∅" is 0.90 for flexure and 0.85 for diagonal tension, bond, and anchorage.

For principle compression and tension, the liner stresses are maintained below 0.95 specified minimum yield at normal operating temperature (i.e., ∅=0.95). For shear, the liner stresses are maintained below 0.6 yield.

The liner is designed to assure that no strains greater than the strain at the guaranteed yield point will occur at the factored loads. In regions of local stress concentrations or stresses due to localized secondary load effects, the liner is permitted to yield but the maximum liner strain is limited to 0.5-percent. Sufficient anchorage is provided to ensure elastic stability of the liner. The basic design concept for the liner stud anchorage is the ductility of the anchorage that assures stud failure due to shear, tension or bending stress without the stud connection causing failure or tear of the liner plate. References 2 and 3 provide information on design of stud connection. The studs in the 0.50-in. plate are installed on 24-in. horizontal and 28-in. vertical grid and in the 0.375-in. plate on a 24-in. horizontal and 14-in. vertical grid. Studs are centered between vertical bars. In the dome, 5-ft by 5-ft panels are anchored in the center by studs and by T−bars at the edges. The 0.50-in. diameter bent welding studs are 9-in. long minimum and 9.50-in. long maximum with a 2-in. 90 degree hook at the end. An arc stud welding process was used on all bent welding studs. The arc stud welding process produces a circular weld around the 0.50-in. diameter stud with a diameter (outside to outside of weld) equal to 0.678-in. and a height equal to 0.157-in. The design considers the possibility of daily stress reversals due to ambient temperature changes for the life of the plant, and fatigue limit of the studs exceeds the design requirements. However, to accommodate possible fatigue failure in the plate-to-stud weldment, the depth of penetration to the liner plate is controlled to avoid impairment of liner integrity.

The boundary conditions in the cylinder are determined by assuming a buckling model (shown in Figures 5.1-17 through 5.1-19) in which the studs form the low points and the center of the panels form the high points of a series of peaks and valleys thus forming a set of panels whose edges represent points of inflection. The analytical procedure used is a simply supported plate under biaxial compression. A Mohr's circle analysis is used to find the normal and shear stresses on this simply-supported plate. The critical buckling stress is derived considering a plate whose length is equal to one-half of the diagonal distance between studs. This critical buckling load is 38.1 ksi for the 0.375-in. liner and 38.4 ksi for the 0.50-in. liner, which is higher than the yield strength of the liner, 32 ksi; therefore, the liner plate will begin to yield before the critical buckling stress is reached, and buckling failure does not control the design. Since shear reduces the stability of a plate subjected to compressive stresses, critical shear is considered and it was found that critical buckling is controlled by normal stresses rather than shear stresses. This is determined by considering the magnitude of both the normal and the shear stresses on the panel. The magnitude of the shear is so low that it shows no effect on the previously stated critical buckling stresses.

In the dome the liner will be considered clamped at the stiffeners forming a 5-ft by 5-ft grid panel pattern. The center of each panel is fixed by a stud. Assuming points of inflection at the one-quarter point a distance of 1-ft 3-in. occurs between points of simple support. The critical buckling load is 58.1 ksi, which is also higher than the yield strength of the liner.

At maximum strain in the liner, the studs will not fail. This maximum strain due to an unbalanced load would occur in a panel adjacent to a buckled panel. Since this adjacent stud will not fail, no zipper effect will occur and massive buckling of the liner and mass failure of anchors is not credible.

The anchorages can fail by failure of the studs in shear or tension, by studs pulling out from the concrete, or by studs separating from the liner plate. The most likely mode of failure is by tensile failure of the stud. The anchors are designed so that failure occurs in the anchor rather than the plate, thereby ensuring that the leaktight integrity of the containment liner will be maintained.

If failure should develop, it would be a random stud failure due to poor workmanship during stud attachment. This failure would not impair the liner integrity nor would it cause progressive failure.

The anchor must resist tensile and shearing loads. Tests have indicated that the lateral load needed to prevent column buckling is 1-percent of the axial yield load. Conservatively doubling this value to account for uncertain field conditions, a value of 2-percent is used.[4] The total load per plate would be 24-in. x 0.50-in. x 32,000 psi = 384,000 lb. Therefore, the tensile load per anchor is 384,000 lb x 0.02 = 7680 lb, which yields a stress of 7680/0.2 = 38,400 psi.

This compares with a yield value of 50,000 psi and a tensile strength of 60,000 psi in the studs. This does not consider the internal pressure, which provides further stability against buckling.

The shear load on the anchor is due to the strain in the liner. Assuming the liner approaches its yield strain of 0.1-percent, the anchor deflection would be 28-in. x .001 = .028-in. Tests on the stud anchor have shown a maximum deflection of about 0.1-in. can be tolerated before failure of the stud.

5.1.2.5 Missile Protection

Except for the upper portions of the steam generators and the Pressurizer, the high pressure reactor coolant system equipment is surrounded by the 3-ft concrete shield wall enclosing the reactor coolant loop and by the 2-ft concrete operating floor.

In 1989, the NRC approved changes to the design basis with respect to dynamic effects of postulated primary loop ruptures, as discussed in Section 4.1.2.4.

A structure is provided over the control rod drive mechanism to block any missiles generated from fracture of the mechanisms.

Systems containing hot pressurized fluids and that might affect the engineered safeguards components have been carefully checked against the possibility of being sources of missiles. The general criterion adopted has been to take provision, when necessary, against the generation of missiles rather than allow missile formation and try to contain their effects.

Once the design requirement that the above systems are not to be sources of missiles has been set forth, identification of potential deficiencies and generation of adequate fixes took place through the quality assurance program.

The following examples illustrate how this approach has been implemented.

5.1.2.5.1 Valves

Valves installed in the nuclear steam supply system have stems with back seat. This rules out the probability of ejecting valve stems; even if it were assumed that the stem threads fail, analysis shows that the back seat or the upset end cannot penetrate the bonnet and thereby become a missile. Additional interference is encountered with air- and motor-operated valves.

Valves with nominal diameter larger than 2-in. have been designed against bonnet-body connection failure and subsequent bonnet ejection by means of:

(1) following the EPRI recommendations[12] regarding bolting practices;
(2) using the design practice of ASME Section VIII for flange design; and
(3) by controlling the pre-load during the bonnet body connection stud tightening process.

The pressure-containing parts except the flange and studs are designed per criteria established by USAS B16.5. Flanges and studs are designed in accordance with ASME Section VIII. Piping and flange materials of construction are procured per ASTM A182, F316, or A351, GR CF8M.

Stud and nut material is ASTM A193-B7 and A194-2H. The proper stud torquing procedures and the use of a torque wrench, with indication of the applied torque, limit the stress of the studs to the allowable limits established in the EPRI Good Bolting Practices Reference Manual (NP-5067).[12] This stress level is far below the material yield, i.e., about 105,000 psi. The complete valves are hydro-tested per USAS B16.5 (1500 lb USAS valves are hydro-tested to 5400 psi). The cast stainless steel bodies and bonnets are radiographed and dye penetrant tested to verify soundness.

Valves with nominal diameter of 2-in. or smaller are forged and generally have screwed bonnet with canopy seal. The canopy seal is the pressure boundary while the bonnet threads are designed to withstand the hydrostatic end force. The pressure containing parts are designed per criteria established by the USAS B16.5 specification.

5.1.2.5.2 Reactor Coolant Pump Flywheel

The reactor coolant pump flywheel is not considered to be a credible source of missiles because of conservative design and care in manufacture and inspection. The flywheel material is ASTM A-533 having an nil-ductility transition temperature less than 10°F. The design results in a primary stress less than 50-percent of the material yield strength at operating speed. The flywheel was subjected to 100-percent volumetric ultrasonic inspection, which is repeated at intervals during plant life. The finished machined bore is subjected to either magnetic particle or liquid penetrant examination. The design overspeed of the pump is 125-percent. The maximum pump overspeed on loss of external load is 112-percent. For an additional discussion of integrity of the reactor coolant pump flywheel, see Section 4.2.2.

5.1.2.6 Quality Control

To ensure a high degree of confidence in plant design, construction, workmanship, materials, and performance, a quality control program has been in effect for this project in which the following principal organizations have their respective responsibilities:

1. Consolidated Edison Company of New York, Inc. as initial owner and operator of the plant.

2. Westinghouse Electric Corporation as the turnkey plant contractor and supplier of major equipment.

3. United Engineers and Constructors Inc. as architect-engineer, construction manager, and constructor.

The function and responsibility in the quality control program of each of the above organizations is as follows:

5.1.2.6.1 Consolidated Edison Company of New York, Inc. (Con Edison) – Initial Licensee

A qualified field representative was assigned to the field during the construction period. His responsibilities included continuous inspection of the construction of the containment building to ensure that all materials used and work performed was strictly in accordance with the plans and specifications. The Con Edison representative, through instructions received from the home office, had the power to stop the construction until any discrepancies were corrected and the work once more was in compliance with the specifications and plans.

The Con Edison representative was in constant communication and consultation with the construction superintendent in matters regarding quality control. In addition, personnel from U.S. Testing Laboratories were assigned to this project to monitor the inspection of the construction and obtain samples of the materials for testing.

5.1.2.6.2 Westinghouse Electric Corporation

For the assurance of plant integrity and quality, Westinghouse performed the following functions regarding the containment building:

1. Reviewed and approved the containment design criteria, material specifications and detail design concepts before they were released for construction. This work was done by qualified structural engineers at the company's home office.

2. Reviewed the construction and inspection methods employed by United Engineers and Constructors Inc.

Westinghouse Pressurized Water Reactor Division, Nuclear Power Services Group had a field quality assurance representative in residence during the construction period. His function was the same as the Con Edison representative mentioned above. He reported discrepancies to the Westinghouse Construction and Services resident engineer who had the authority to stop the work until the discrepancy was resolved.

In addition to this, he audited the construction files, and verified that records were complete, accurate, and adequate for quality assurance.

Nuclear Power Service Headquarters quality assurance engineers also made trips to the site to audit, monitor, and review the project with regard to site quality assurance. Construction practices were observed for conformance to codes, specifications, and approved procedures.

5.1.2.6.3 United Engineers and Constructors Inc.

The responsibilities of United Engineers and Constructors Inc. in the quality control of the containment building were as follows:

1. They inspected all materials delivered to the job site, and examined the suppliers' certified test reports of physical and chemical properties for those components furnished by them.

2. They inspected fabrication of major components of the containment structure in the shop. Trip reports are available at the site.

3. They maintained an adequate force of qualified supervisory personnel at all times.

4. They supervised and were fully responsible for the quality of work performed by their subcontractors and for the craft labor employed and supervised by them.

5. They maintained as part of their field engineering force, qualified personnel who performed a thorough inspection of each construction operation.

No changes in design or specifications were allowed without the approval of the engineer in charge of design.

5.1.3 Containment Stress Analysis

5.1.3.1 General

The structural design of the containment meets the requirements established by 1961 edition of "The State Building and Construction Code for the State of New York" so far as these provisions are applicable. All concrete structures have been designed, detailed, and constructed in accordance with the provisions of "Building Code Requirements for Reinforced Concrete" (ACI 318-63) so far as these provisions are applicable.

5.1.3.2 Method of Analysis

Basically three separate structural components have been analyzed, each in equilibrium with loads applied to it and with constraints occurring at the juncture of the structures. The three components are:

1. The 135-ft ID hemispherical dome.
2. The 135-ft ID cylinder.
3. The base slab.

Mathematically, the dome and cylinder have been treated as thin-walled shell structures, which results in a membrane analysis. Since the thickness of the dome and cylinder is small in comparison with the radius of curvature (1/20 and 1/15) and there are no discontinuities such as sharp bends in the meridonal curves, the stresses due to pressure and wind or earthquake are calculated by assuming that they are uniformly distributed across the thickness.

Since the concrete is not assumed to resist any tensile or shear forces, radial shear reinforcing has been introduced in the lower portion of the wall in the form of hooked diagonal stirrups and diagonally bent bars as shown in Figure 5.1-1. Diagonal shear reinforcing, at 45° and 135° to the circumferential direction, are placed in the center of the cylinder wall for the full height of the wall and a distance above the springline into the dome to resist earthquake shears. The diagonal bars are discontinued in the upper area of the dome (beyond about 30 degrees above the springline), where the seismic shears are small and are carried by the dome reinforcing steel lying in the plane of principal tension.

The base slab has been treated as a flat circular plate supported on a rigid nonyielding foundation.

The limiting cases in the design of the wall for discontinuity moments and shears were considered. One case considered an uncracked wall and the other considered a cracked wall with the steel acting as a spring constant. The value of μ_c varied from zero in the cracked case to .14 in the uncracked case. In the uncracked case, variations in E_c will have no effect on the answer since E_c appears in both the numerator and the denominator of the stiffness formulation. For the above variation in E_c and μ_c, the values of discontinuity moment and shear vary by 14-percent and 7-percent respectively at the base. These are the maximum deviations of the wall forces since the wall will actually vary from uncracked to cracked with an increase in containment height rather than be cracked or uncracked for the total height.

In the area of thermal stress, the entire wall section will be cracked and no variation in E_c or μ_c need be considered. The liner stresses depend on the strains of the reinforcing steel and are not related to the concrete properties.

Shrinkage and creep effects will be relieved by cracking during the pressure test and will not be included in accident design considerations.

The finite element computer program has the capabilities of taking into account variations in μ_c and E_c and axisymmetric loads. However, it is not necessary to take into account the variations in μ_c and E_c for the reasons stated above.

The computer program used to study the general behavior of the structure and to generate boundary conditions was the axisymmetric shell structure program. This computer program, developed by Franklin Institute Research Laboratories, is designed to handle arbitrarily shaped shells of revolution subjected to axisymmetric as well as nonaxisymmetric loadings. The method of analysis consists of subdividing the shell into elements having continuous meridians with continuous first and second derivatives so that the first and second fundamental forms of the resulting shell elements are continuous throughout the element. By expanding the dependent variables in Fourier series in the circumferential direction, and assigning unspecified functions for the meridional variation, the independent variables are separated and a system of ordinary differential equations results for the dependent variables in terms of the meridional independent variable. Particular and complementary solutions of these ordinary differential equations are then found for each of the elements and each of the circumferential harmonics individually. The matching of the elements is achieved by writing the required boundary conditions.

The idealized section used with the axisymmetric shell structure program consists of five layers whose moment of inertia is equal to that of the actual section. The wall section is considered as cracked with the reinforcing carrying all loads.

A finite element program, with the capability to incorporate thermal loads, was used to analyze the containment shell considering the effect of the equipment hatch opening.

The shell was idealized into 10 layers with alternate layers of steel and concrete. Section 5.1.3.10 provides more information on the finite element analysis.

The computer program can handle the loads in the form of either surface traction or edge loads or both.

Analysis of the liner is presented in the Containment Liner Stress Analysis Report. The report also contains a description of analytical procedures arriving at forces, shears, and moments in the structural shell.

5.1.3.3 Dome Analysis

The analysis of the hemispherical dome has been performed by the super-position of membrane forces resulting from gravity, accident pressure, and accident thermal loads. In addition, earthquake or wind loading create both direct and shear stresses in the dome, and the operating temperature of the liner creates tension and compression. All of the combined direct stresses are developed in the reinforcing steel encased in the concrete. In the upper area of the dome (about 30 degrees above the springline), where the seismic shears are small, seismic shears are carried by dome reinforcing steel lying in the plane of the principal tension. The dome reinforcing is spliced to the vertical steel in the cylindrical concrete wall, so that a continuity between the dome and the cylinder is realized. See Figure 5.1-20 for a section of wall, dome and for reinforcing in the dome.

5.1.3.4 Cylinder Analysis

The analysis of the cylinder is by superposition of membrane forces resulting from gravity, pressure and thermal loads, overturning due to earthquake or wind and shears due to earthquake or wind. The concrete has been reinforced circumferentially using steel hoops and vertically by straight bars. Diagonal bars have been placed to resist the horizontal and vertical shears due to earthquake or wind. The required capacity of the diagonal bars has been designed so that the horizontal component per foot of the diagonals is equal to the maximum value of shear flow. A check was made to ensure that no net compressive force results in the diagonal bars because of the combination of seismic shear load and internal pressure load. Although, in the cylinder, the liner has some capacity available to resist the seismic shears, no credit is taken for this capacity.

For all of the cylinder and the lower areas of the dome, the diagonal reinforcing has been designed to accommodate all seismic shears. No credit has been taken for the dowel action of the vertical and horizontal bars in resisting seismic shear.

Only in the upper area of the dome (beyond about 30 degrees above the spring line) where the seismic shears are small is the liner counted on to resist shear. For all of the cylinder and the lower areas of the dome, the diagonal reinforcing has been designed to accommodate all seismic shears. No credit has been taken for the dowel action of the vertical and horizontal bars in resisting seismic shear.

5.1.3.5 Base Mat Analysis

The base slab was treated as a flat circular plate supported on a rigid non-yielding foundation. For loads applied uniformly around the slab, the analysis considers a 1-ft wide beam fixed at a point where the vertical shear is equal to zero. This is the point where the downward pressure on the mat and the dead weight overcome the uplift at the containment wall base mat juncture from pressure and earthquake loadings. Radial and circumferential reinforcing is provided at the top and bottom of the mat to resist moments in the areas where uplift occurs. Temperature steel was added in other areas to meet requirements of the (ACI-318) Code. Diagonal tension reinforcement was added to meet requirements of ACI-318 Code. See Figure 5.1-23 for base slab reinforcing detail.

Moments and shears were calculated by writing equations for moment and shear in terms of X using the containment wall-base slab juncture as the origin with X increasing toward the center of the containment building. The point along the circumference of the containment wall chosen as the end of the beam is a point where the maximum tension from the earthquake will exist. Since the containment structure is considered a beam in all earthquake analyses, the maximum uplift for which the mat is designed will occur at only one point on the circumference and will represent the worst possible uplift on the mat.

All stresses were calculated using Part IV-B Structural Analysis and Proportioning of Members - Ultimate Strength Design of the Building Codes Requirements for Reinforced Concrete (ACI–318-63). No rebar stresses exceed 0.90 f_y.

A gradient with an operating temperature of 120°F inside the containment and a 50°F temperature at the mat-rock interface was considered and stresses were negligible. Ambient accident temperatures have no appreciable effect on the base slab. The maximum operating temperature of the containment is 130°F. The effect of elevated operating temperature on the structural elements was evaluated in 1987 and was found acceptable.

It is not possible to show that the design on nonyielding rock is more conservative than assuming the rock to be elastic. However, due to the installation of temperature reinforcing, the design is conservative. Reinforcing and concrete stresses are very low when considering the rock to be elastic.

To substantiate the above statement, the following studies were performed:

1. The foundation modulus were determined using the expression:[15]

$$k_z = \frac{4Gr_o}{1-\mu}$$

where:

k_z = The vertical spring constant of a circular base supported on an elastic foundation

$$G = \frac{E}{2\ (1+\mu)}$$

r_o = Radius of Foundation

μ = Poisson's Ratio

To obtain the foundation modulus, k_z is divided by the area of the circular base to yield

$$k_o = \frac{k_z}{A} \times \frac{4\ G}{\pi r_o \left(1 - \mu\right)}$$

Substituting for G

$$k_o = \frac{2\,E}{\pi\, r_o (1 - \mu^2)}$$

2. The first case examined was that of a rectangular strip loaded with 1.5 times design accident pressure plus dead load using conservative properties for the Dolomitic limestone:[7,14]

$E = 6.0 \times 10^6$ psi

$\mu = 0$

Applying these values

$k_o = 4370$ lbs/in.^{3}

The "characteristic" λ is defined as:[6]

$$\lambda = \left[\frac{k}{4\ EI}\right]^{1/4}$$

Where:

E is the modulus of elasticity of the structural base (concrete),

I is the moment of inertia of the structural base,

$k = k_o$ b , (b = width of base)

using base properties

$\lambda = 7.56 \times 10^{-3}$-in.$^{-1}$

Where $\lambda\ell > \pi$ beams may be considered as infinite in length.[6]

Taking the length of beam as being the base diameter

$\lambda\ell = 13.1 > \pi$

The beam was then analyzed as a beam of unlimited length loaded over an area equal to the base diameter with an 80 psi uniform load.

The solution to this problem gives

$$y_C = \frac{q}{2k}\left(2 - D_{\lambda a} - D_{\lambda b}\right)$$

$$M_C = \frac{q}{4\lambda^2}\left(B_{\lambda a} + B_{\lambda b}\right)$$

$$Qc = \frac{q}{4\lambda}\left(C_{\lambda a} - C_{\lambda b}\right)$$

where

y_C is deflection of point being considered

M_C is the moment at point being considered

Q_C is shear at point being considered

q is the uniform load

a is the distance from point under consideration to end of load

b is distance from point under consideration to other end of load.

$B_{\lambda x} = e^{-\lambda x} \sin \lambda x$

$C_{\lambda x} = e^{-\lambda x}(\cos \lambda x - \sin \lambda x)$

$D = e^{-\lambda x} \cos \lambda x$

Maximum moment occurs at mid-point of load and is equal to 352-in.-lbs/in.

For the area of the mat where there is only temperature reinforcing, the maximum moment would cause a stress of 30 psi in the reinforcing.

The maximum shear would occur at the ends and is equal to 2.64 kips/in. This shear would cause a shear stress in an unreinforced concrete section of 26.4 psi.

3. A second case examined was for the foundation material being less rigid than the concrete base. The model was the same for the first case:

Assumed $E_{rock} = 2.6 \times 10^6$ psi

$\mu = 0$

For this case, the following were determined:

$k_o = 1890$ lb/in.^{3}

$\lambda = 6.2 \times 10^{-3}$-in.$^{-1}$

M_{max} = 3.66-in.-kips/in.

Q_{max} = 3.23 kips/in.

S_{rebar} = 312 psi

ν_{conc} = 32.3 psi

As a final study, the maximum deflection as calculated in the first case was imposed as a settlement of the base mat for the outer portion and a section of the mat was analyzed for this settlement. A 30-ft section was used with fixity at the reactor pit, the remainder cantilevered from the pit.

The resulting moment and shear are as follows:

M = 142-in.-kips/in.

q = 396 lbs

resulting in a rebar stress of 12.2 ksi and a shear stress of 4.0 psi.

From the above, it can be seen that the assumption that a foundation on rock is a rigid unyielding foundation is a valid assumption and that temperature reinforcing provides much greater resistance than required to accommodate the effects of any elastic deformation of the subgrade.

5.1.3.6 Analysis of Liner and Reinforcing Steel

Approximately 67-percent of the inclined bars, provided to resist radial shear at the base of the containment wall, are secondary vertical bars, which are inside the primary vertical bars on the outside face and inside face of the wall. These bars are continuous and are bent across the wall where reinforcing is required to resist the radial shear. The remaining 33-percent of the required steel area is provided by stirrups that are hooked around the vertical bars by means of a 90 degree hook. Only one-third of the shear reinforcing at a particular elevation is made up of these hooked bars, which occur at four elevations up the wall. See Figure 4.16 of the Containment Design Report.

Since the stud anchors are hooked around reinforcing bars, concrete stresses for pull out loads are negligible. For high shear loads, which would be caused if a stud anchor should fail or be missing, local crushing of the concrete occurs; however, integrity of the anchor and liner plate is not impaired. See Figures 5.1-21 and 5.1-22.

The lowest elevation at which these hooked bars are used is at a point where only 65-percent of the maximum shear at the base is present. The remaining three levels are in regions where the shear is less than 25-percent of maximum base shear. Since the large majority of the shear is resisted by continuous vertical bars, a minimal amount of load must be transmitted to the vertical bars. The hooked stirrups will mechanically transmit the small amount of shear, which they carry. The main function of the stirrups is to contain the formation of the diagonal tension crack. The mechanical anchorage of the stirrups is sufficient for this purpose.

There are no significant structural loadings, which must be transferred through the liner such as those required for crane brackets or machinery equipment mounts. Miscellaneous spray system piping, instrumentation, conduit, and insulation, which are attached to the liner can be supported by the free-standing liner without inducing significant stresses in the liner or liner anchorage.

Liner stress is imposed on the cylindrical penetration as a circular uniform load acting around the circumference of the penetration. The liner plate is locally thickened at the penetrations to take care of additional stresses.

The liner can accommodate any shear it will see due to thermal expansion or earthquake.

An investigation was made on the thermal effects, based on the conservative assumptions that the base mat was fully fixed against any thermal movement thereby restraining the liner from movement. The 3-ft fill slab was then subjected to thermal growth. No excessive forces were introduced into the liner and the welds on the test channels were found to be sufficient to prevent any shear failure of the test channels from the liner due to movement of the 3-ft fill mat.

Seismic shear of the interior concrete is resisted by the keying action of the reactor pit and the sump for the recirculation pumps in addition to the weld channels. Considerable resistance is also provided by friction between the liner and the 3-ft slab.

Jet forces cannot remove the liner panels since the forces will be compressing the insulation panels against the liner and exterior wall. The panels are anchored to the liner with 3/16-in. diameter stainless steel studs. The consequence of an insulation panel being displaced from the liner during or as a consequence of an accident is that the exposed liner would tend to expand. The unequal strain between the exposed and unexposed portions of the liner causes a shear load on the liner anchor, and a local yielding in compression of the exposed portion of the liner. The liner anchor stud has the capacity to accommodate much greater strains than would be experienced at yield strain in the liner.

5.1.3.7 Containment Interior Structure

The interior structure may be separated into five main structural components. They are:

1. 3-ft thick fill slab.
2. 3-ft thick crane wall.
3. 4-ft to 6-ft thick refueling canal.
4. 2-ft thick operating floor slab.
5. Primary shield wall.

The method of design, stress analysis, critical stresses and locations are as follows:

1. 3-ft thick fill slab - The controlling loads on the 3-ft slab are the reactions are from the primary equipment supports due to various postulated pipe breaks. The slab was designed as a series of radial beams running under the equipment supports and spanning between the reactor support wall and the crane wall. Stresses in reinforcing were limited to 0.9 fy. Maximum stresses occur immediately below the primary equipment supports.

2. <u>3-ft thick crane wall</u> - The crane wall was designed for a 7 psi differential pressure occurring immediately after a primary pipe break and prior to pressure equalization.

   Although the stress levels associated with this pressure differential were sufficiently low to establish that the concrete could resist the pressure loading, sufficient reinforcing was provided to resist all membrane forces without any contribution from the concrete. Stresses were limited to 0.9 fy. The membrane hoop stress was 33 ksi and the axial vertical rebar stress was 14.3 ksi.

A two dimensional finite element analysis was performed to determine the effect of the jet forces associated with the pipe break on the crane wall.

The jet force associated with a pipe break has been based on the static force PA where P is the primary system operating pressure and A is the cross sectional area of the coolant pipe. The analysis indicated that in local areas (at the application of the force) yielding of the crane wall rebar will occur. The load was assumed to act at the mid-height of the wall, thus causing maximum bending moment. The ability of the wall to support the dead load of the crane was checked, considering the yielded area indicated by the computer analysis as unable to carry load. A beam 12-ft long and 5-ft deep (the underside of the operating floor to the top of the potential yield portion of the crane wall) was found to provide more than twice the ultimate capacity required. This analysis was very conservative for three reasons:

1. A jet force load at this location would cause little yielding since it is not located at mid span.
2. The haunch at the underside of the operating floor was not considered.
3. The membrane effect of the circular crane wall was not taken into account.

Further stability of the crane wall was demonstrated by determining the ultimate failure load by means of a yield line analysis. This analysis indicated that the structure has the capacity, through strain energy of structural response, to resist the uniform jet force load of 1500 kips or 975 kips with the 7 psi pressure differential without failure.

The containment internal concrete is essentially rigid; (fundamental frequency 18.6 cps) therefore, seismic loads were calculated using the maximum ground acceleration (0.15g).

The crane wall was initially considered as a cantilever beam with a frequency of approximately 13 cps and the base shear was determined by the response spectrum approach. The base shear was distributed to the individual nodes by the formula:

$$Fx = \frac{W_x h_x V}{\Sigma W h}$$

Where

V = base shear
W_x = weight of node under consideration
h_x = distance from base to section under consideration.
3 Wh = Summation of the product of weights and heights of all nodes

The moment at the base was determined and the uplift calculated by considering a circular ring of thickness equal to the area of steel per in. This maximum uplift, which occurs at one point at the base of the structure stresses the rebar to 5.2 ksi.

The crane wall was also designed to resist steam and feed water pipe break reactions of 340 kips and 200 kips where supports are connected to the wall. The extra steel provided for pipe break loads is available in the form of steel buttresses to resist pressure, jet force, and seismic loads; however, it was not considered in the analysis.

3. 4-ft to 6-ft thick refueling canal - The refueling canal was designed for the 7 psi pressure differential. The wall resists the pressure by spanning vertically between the refueling floor and the operating floor. Stresses were limited to 0.9 fy.

An analysis was performed to check the effects of the jet force load the cross section was found to be sufficient to provide stability. A yield line analysis was performed and provided the basis for the above.

The seismic load was determined by the same procedure used for the crane wall. The average load in kips/ft was distributed over the wall and the vertical span was conservatively assumed to carry the entire load. The resulting bending moment was found to be well within the capacity of the wall.

4. 2-ft thick operating floor slab - Because of the many openings in the floor for equipment, the floor was designed as a series of beams. Principal loadings were D.L. + 500 psf live load and 7 psi upward pressure differential + D.L. The first loading (D.L. + 500 psf live load) was designed in accordance with Part IV-B of ACI 318. Stresses for the pressure differential case were limited to 0.9 fy.

   The operating floor was investigated. There appears to be very little area of the operating floor, which could be reached by the expanding jet of water from a break in the reactor coolant system. The jet will be greatly dispersed in the distance between the primary coolant piping and the underside of the operating floor. The only area of the floor, which could be struck by a jet spans between areas of the floor heavily reinforced as beams. The span cross section consists of a T-beam with the 2-ft thick floor acting as the flange and the 7-ft high biological shielding wall as the web. This section can resist the jet force load within 0.9 fy stress limit on the rebar.

5. Primary Shield Wall - This was designed for two loading conditions due to a split in the reactor. The stress in the reinforcing was limited to the tensile strength of the bars. The first load considered was a 1-ft wide longitudinal split along the length of the reactor. The vessel is assumed accelerated through a 6-in. distance against the support wall by the jet force caused by a 2200 psi pressure acting through a 26.4-ft long by 1-ft wide longitudinal vessel rupture, which results in an impact load of 650 k/ft. This load is imposed by considering an impact factor of two. The maximum rebar stress is 69.5 ksi. The second load considered a pressure buildup of 1000 psi inside the pit due to release of reactor contents. This produces a rebar stress of 86 ksi. The rebar used is ASTM A 432 with specified yield of 60 ksi and ultimate tensile strength of 90 ksi.

To protect the containment base liner, an average of 2-ft of concrete above the containment liner plus a 1-in. liner plate embedded on top of the concrete was provided at the bottom of the containment reactor cavity pit. Below the containment liner plate is 4.5-ft of structural concrete poured on rock.

Temperature differential conditions as a result of a LOCA are considered to be of such short duration that the effects were not used in the design of interior structures for stress analysis. A sketch of the design conditions is given in Figure 5.1-24.

During normal operations, the only significant transient temperature gradients occur during startup. The minimum containment internal temperature is limited to 50°F. The maximum operating containment internal temperature is 130°F. Forced movement of containment air is used to limit the concrete temperature surrounding the reactor vessel. This forced air movement of the containment air as well as normal convection and radiation is expected to limit the concrete temperature differentials in the range of 5°F to 10°F. To demonstrate the large margin available in the concrete crane wall and the primary shield wall, a conservative assumption of a 30°F temperature gradient has been evaluated. The evaluation included the gradient effect through the crane wall, the 6-ft thick portion of the primary shield wall below the reactor coolant pipe nozzle, the 5-ft thick portion of the primary shield wall where the nozzles penetrate the wall, and the 4-ft thick wall above the shield wall.

The maximum rebar stress was found to be 4500 psi and occurs in the vertical rebar in the crane wall. The maximum compressive concrete stress was found to be 226 psi and occurs in the hoop direction in the 5-ft portion of the primary shield wall. These stresses are approximately 20-percent of the allowable working stress values and will have no significant effect on the design adequacy of the structures analyzed.

5.1.3.8 Pressure Stresses

5.1.3.8.1 Accident Pressure

Pressure effects on the containment structure may be divided into two types:
(1) membrane stresses and (2) discontinuity stresses.

1. For membrane stress analysis, the dome and cylinder are treated as thin-walled shell structures. (The thickness to radius ratio for the dome is 1/20 and the cylinder 1/15. These ratios are smaller than the 1/10 criterion for thin-walled shell analysis.[8] Membrane forces are resisted by steel reinforcing.

2. Discontinuity stresses occur at the juncture of the cylinder and the mat and the juncture of the cylinder and dome. Discontinuity effects are determined as follows:

   a. The radial growth of the shell is computed based on membrane stress in the reinforcing and liner.

b. The flexural rigidity of the meridional wall section is determined based on a cracked section analysis in accordance with conventional reinforced concrete design techniques.

c. Moments and shears are calculated based on having consistent deformation for the two elements at the point of discontinuity.

Discontinuity effects at the spring line are very slight due to the small difference in radial growth between the dome and cylinder. Since the circumferential reinforcing in the dome and cylinder vary, stresses and therefore deformations are essentially equal.

The mat is considered as offering complete fixity; no credit is taken for the liner at the base in resisting moments since at the point of maximum shear the bond between the liner and concrete is insufficient to transmit complementary beam shear. A slip surface between the concrete and liner is formed and the liner is subjected to membrane forces only.

The 9-ft thick mat is subjected to the following due to pressure inside the containment building:

1. Uplift at the juncture with the wall.
2. Moment and shear due to discontinuity effects with the wall.
3. Downward pressure loading due to internal pressure.

The 9-ft mat is designed to accommodate the flexural effects of these loads. At the crane wall, the mat is founded on the unyielding rock and further pressure loads are transmitted through bearing directly into the rock.

Resistance to these loads is based on a cracked concrete section. No credit is taken for the liner for the same reasons given for the wall.

Discontinuity shears in both the cylinder and mat are resisted by either bent bars or stirrups.

In the outer portions of the base mat, the slab is raised off of the rigid foundation under accident loadings; thus no frictional resistance can be offered by the rigid foundation. Where the uplift is overcome, the only load of any consequence, which must be resisted by the mat is the radial tension. The restraint, which is imposed by the rigid foundation on the bottom portion of the base mat, effectively eliminates all radial tension in the mat. However, for conservatism this restraint has been neglected in the analysis of the mat for radial tension. The hoop and radial reinforcing supplied as temperature reinforcing is more than adequate for this purpose.

5.1.3.8.2 Soil Pressure

Portions of the containment structure are subjected to the effects of backfill bearing against the containment wall. The effects on the structure are:

1. Shear and overturning effects due to seismic response and interaction between the soil and structure.

2. Discontinuity effects caused by the soil restraining deformation of the structure under accident pressures.

To determine the shear and overturning effects two limiting cases were investigated. The first was the case where the structure and soil move out of phase. It was assumed that the structure was subjected to the passive pressure of the soil with the mass of soil, within the shear failure envelope, accelerated against the structure with ground acceleration. In the second case the soil and structure move in phase. For this case it was assumed that the structure was subjected to the active pressure of the soil with the mass of soil, within the shear failure envelope, accelerated with the structure at ground acceleration.

These loads were then treated as external loads on the structure. See Section 3.1.5 of the Containment Design Report for additional information.

To determine the discontinuity effects caused by soil restraint, the structure was analyzed for the passive pressure case. The restraint of the deformation of the structure due to the soil was calculated. Vertical and circumferential bending moments due to this restraint were then determined. Reinforcing bar stresses were calculated and found to be minor. This analysis was then verified by a finite element analysis.

In this analysis, full contribution of the backfill was assumed. During the course of construction it became necessary to build a retaining wall in a substantial area of the backfill, to facilitate construction. The retaining wall extends over 50-ft in plan and includes all of the high fill points assumed in the analysis and design. It can therefore be concluded that the analysis was conservative in that the backfill effects on the completed structure would be only a fraction of that assumed in the original design.

5.1.3.9 Thermal Stresses

Temperature effects on the containment structure may be divided into two separate considerations: one effect is due to a thermal gradient through the wall, the other is caused by the rapid temperature rise of the liner under accident conditions. The reinforced-concrete wall restrains the liner from growing, resulting in compression in the liner and additional tension in the reinforcing.

1. Calculation of gradient stresses is based on method of analysis outlined in ACI 505-54, "Specification for the Design and Construction of Reinforced Concrete Chimneys."[9] The gradient used is linear with 120°F on the inside and 0°F exterior concrete temperature (-5°F ambient). The maximum operating temperature of the containment is 130°F. The effect of elevated operating temperature (up to 150°F) on the structural elements was evaluated in 1987 and was found to be acceptable.

   The ACI method assumes a cracked section in which the concrete carries no tension. The neutral surface (surface at which no thermal stress exists) is determined. Stresses in the liner and reinforcing are calculated based on the assumption that there is no distortion of the wall; i.e., variation of strain through the wall thickness is linear.

2. To determine the effects due to rapid rise in liner temperature, there are two basic assumptions made. The first is that the effects are internal in nature; i.e., the compressive force in the liner is balanced by a tensile force in the reinforcing. The second is that there is no distortion of the wall.

Because temperature effects are internal in nature and do not affect the overall tensile load carrying capability of the structure, local yielding of reinforcing under accident conditions is acceptable.

The temperature gradient through the wall is essentially linear on both the insulated and uninsulated portions and is a function of the operating temperature internally and the average ambient temperature externally. Accident temperatures mainly affect the liner, rather than the concrete and reinforcing bars, due to the insulating properties of the concrete. By the time the temperature of the concrete adjacent to the liner begins to rise significantly, the internal pressure and temperature in the containment shell due to maximum thermal gradient will not influence the capacity of the structure to resist the other forces. Temperature effects induce stresses in the structure, which are internal in nature; tension outside and compression in the inside of the shell such that the resultant force is zero. Loading combinations concurrent with these temperature effects may cause local stresses in the outside horizontal and vertical bars to reach yield; however, as local yielding is reached, any further load is transferred to the unyielded elements. At the full yield condition, the magnitude of final load resisted across a horizontal and vertical section remains identical to that which would be carried if the temperature effects were not considered. Thus, the overall carrying capacity of the structure and the factor of safety of the structural elements are not affected.

5.1.3.10 Analysis of Openings

The methods followed in design of large openings are described in Section 3.4 of the Containment Design Report (CDR). Included are descriptions of the safety factors used in design. Sample calculations are provided, listing all the criteria and analyzing the effects of all pertinent factors, such as cracking. Also addressed in the CDR is how the existence of biaxial tension in concrete (cracking) has been taken care of in the design, and how the normal and shear stresses due to axial load, two-directional bending, two-directional shear, and torsion are combined. Additionally, the criteria for the design of the thickened part of the wall around the openings is stated.

The methods used to check the design of the thickened stiff part of the shell around large openings and its effect on the shell, torsional stresses, and shrinkage considerations are also addressed in Section 3.4 of the Containment Design Report. This section also describes how deformations and forces are handled around the large openings and in the transition zones into the main portion of the structure.

In the cylindrical section of the containment, where there are large openings for access hatchways and penetrations, the reinforcing bars (hoop, vertical and diagonal) are continued without interruption around the openings.

No bar terminates at any openings as illustrated around the penetration in Figure 5.1-1. Also additional bars have been furnished locally to take the stresses developed around large openings. Concrete is locally thickened at the equipment access hatchway area to accommodate all the reinforcing bars required in this area.

A finite element analysis is performed on the large openings. Representation of the structure is by rectangular elements; each element consists of ten layers of orthotropic, elastic material to represent the reinforcement, concrete and the liner. About 1000 degrees of freedom are considered in the model. This analysis is used as a check on the adequacy of the large openings. Results appear in the Containment Design Report.

A finite element analysis of the equipment hatch area indicated local liner plastic deformations during the pressure test. For the order of magnitude and location of these stresses, see Section 3.4 of the Containment Design Report. These deformations have no influence on the structure during the pressure test due to the ductility of the studs and liner plate.

The limiting elastic liner deformations during test pressure will be from tensile stresses. During an accident loading they will be from compressive stresses. Therefore, a relationship between the pressure and accident loads cannot be determined directly. However, the test pressure demonstrates the ductile behavior of the liner.

Since the containment is not subject to accident temperatures during the testing, no direct correlation between test and accident conditions can be made in evaluating thermal stresses at large openings.

The liner is stressed beyond the yield point in very local areas adjacent to the transition from the thickened equipment hatch boss to the cylinder wall. The maximum stress is equal to 39.28 ksi for the 1.5P loading condition. The strain corresponding to this stress (0.17-percent) is below the limits (0.5-percent) stated in Section 2.2.4 of the Containment Design Report. The average liner stress in the cylinder for the 1.5P load combination is approximately -15 ksi in the vertical direction and -2.0 ksi in the horizontal direction.

The maximum rebar stress associated with the 1.5P load combination is approximately 66 ksi in the 4'-6" portion of the containment wall cylinder.

For a complete discussion of liner stresses, see the Containment Liner Stress Analysis Report. For a detailed discussion of liner stresses in the equipment hatch area and further justification of the stresses noted above, see Section 3.4.4 of the Containment Design Report.

All reinforcing is continuous around penetrations. Steps have been taken to ensure that no local crushing of concrete will occur. From Reference 16, it has been determined that in order to prevent local crushing of the concrete, a minimum bend diameter of 31 times the bar diameter is required when the reinforcing is stressed to yield. The angle of bend in the rebar determines the force that will be transmitted to the concrete in the event the bar tries to straighten out due to tension. For this reason most bars are bent at 10 degrees except at large penetrations including the equipment hatch, personnel lock, main steam and feedwater, and air purge penetrations, where the deviation of the bar from its centerline is too large to permit a 10 degree bend. In these cases the bars have been bent at 30 degrees but a tie-back system is used, which prevents a buildup of forces. To prevent this buildup, (in all cases except the equipment hatch penetration), the line of force makes an angle of one-half of the angle of bend, from a horizontal line from the vertical bars and from a vertical line for the horizontal bars and is tangent to the outside of the penetration.

At the personnel and equipment hatches a large void will be carried since, due to the large offset of the bars from their centerline, it will take the bars longer to return to their centerline after passing the penetration. To prevent any cracking and spalling of concrete and to add lost strength to the cross-section, these voids have been filled with added rebar, which achieves bond by means of mechanical anchorage.

The same precautions mentioned above have been taken with the seismic bars. See Figure 5.1-25.

For penetrations between 9-in. and 18-in. in diameter, all the reinforcing bars including primary and secondary vertical bars and diagonal bars have been grouped around the penetrations. Due to the continuity of the bars and the relatively small opening size, no special provisions need be made to resist normal, shear, and bending stresses. The penetrations are keyed into the concrete, thus creating an edge loading, which will put torsion into the wall. The loads are small and the rebar will feel little effects from this torsional loading.

For penetrations greater than 18-in. up to 48-in. in diameter, the bars are continuous. Due to the large angle of bend of these bars, a tie-back system is used, which offers additional resisting strength to shear, bending, and torsional stresses.

5.1.3.11 Seismic and Wind Design

The design of the containment, which is a Class I structure (see Section 1.11), is based on a "response spectrum" approach in the analysis of the dynamic loads imparted by earthquake. The seismic design takes into account the acceleration response spectrum curves as developed by G. Housner. Seismic accelerations have been computed as outlined in TID-7024[10] and Portland Cement Association Publication.[11]

The following damping factors have been used:

	Component	Percent Critical Damping
1.	Containment structure	2.0
2.	Concrete support structure of reactor vessel	2.0
3.	Steel assemblies:	
	a. Bolted or riveted	2.5
	b. Welded	1.0
4.	Vital piping systems	0.5
5.	Concrete structures above ground:	
	a. Shear wall	5.0
	b. Rigid frame	5.0

As indicated in Section 5.1.2.2, ground accelerations used for design purposes are 0.1g applied horizontally and 0.05g applied vertically. The natural period of vibration is computed by the Rayleigh method; in this method, the containment structure is analyzed as a simple cantilever intimately associated with the rock base and with broad base sections of adequate strength to assure full and continued elastic response during seismic motions. Further, both bending and shear deformations are considered.

The structure is divided into sections of equal length and loaded laterally by dead weight of the section and any equipment and live load occurring at the section. Deflections caused by shear and moments are then determined, and the end deflection is given the value $\phi' = 1.0$ with corresponding values determined for other sections. The natural period of vibration for the structure is then determined by setting potential energy equal to kinetic energy and solving for the period.

$$T = 2\Pi \left[\frac{Y_0 \sum \varnothing^2 \, dm}{g \sum \varnothing \, dm} \right]^{1/2}$$

where

Y_0 = maximum actual deflection

$$\phi = \frac{\text{deflection of section under consideration}}{\text{maximum actual deflection}}$$

g = acceleration due to gravity

dm = weight of section under consideration

T = period in sec.

Based on an uncracked concrete section, the period is determined to be 0.241 sec. A more realistic calculation for a cracked section, using reinforcing steel and liner as the resisting elements, yields a period 0.936 sec.

Using the derived period and entering the acceleration spectral curves, Figures 1.11-1 and 1.11-2 of Section 1.11, and applying a 2-percent critical damping, a spectral acceleration for the containment was selected. This value was derived to determine the base shear. The distribution of base shear is a triangular loading assumption.

This assumption yields a load distribution pattern with zero loading at the base to a maximum loading at the spring line of the dome. Above this line, the loading decreases due to a change in section and consequently change in weight. This load distribution allows the determination of shears and moments at any critical section through the containment from which the appropriate unit stresses are obtained.

Seismic shears are resisted by diagonal reinforcing except in the upper areas of the dome. No credit is taken for the reinforcing in compression.

From 30 degrees above the springline, where the seismic shears are small, the shears are carried by dome reinforcing steel lying in the plane of principal tension

A finite element analysis was performed on the basemat using loads determined for the three basic loading conditions specified in the Containment Design Report. Maximum hoop moment caused by lack of symmetry of the seismic loading was found to be 454 in.–kips/in. This compares with a capacity of 690-in.-kips/in. for the in-place hoop reinforcing.

Tornado loads have not been considered in the design of the Indian Point Unit 2 Containment Building; however, similarity in design of Indian Point Unit 3 (where such loads are considered) indicates that seismic reinforcement bars provide a more than adequate mechanism to withstand the torsional effect of Tornado loads.

The torsional effect results from wind striking the containment building at an angle α from the normal, as shown in Figure 5.1-26. The torsional force is due to the component of the wind tangential to the surface of the containment building and is equal to:

$$F_t = AC_D\,(q)\,(\sin \alpha)$$

Where

A = surface area of the containment

C_D = 0.5 from A.S.C.E. Paper 3269 - "Transactions of the A.S.C.E.," Vol. 126 Part II 1961, p. 1165 (coefficient of drag)

q = 0.002558 V^2 (wind pressure)

α = 45 degrees

This assumption is conservative in that the actual tangential force would be the result of skin friction and the effects would be negligible.

This component of torsional force is computed from a direct wind loading as based on A.S.C.E. Paper 3269.

Torsional shear is a maximum at the juncture of the walls and base slab and varies to zero at the top of the dome.

The torsional effect can be converted to a shear per lineal foot around the circumference of the containment by distributing the shear over the circumference of the seismic reinforcing.

The seismic bars provide a more than adequate mechanism to withstand this torsional effect. The maximum stress in the bars under this loading is 17 ksi. See Figure 5.1-26.

5.1.3.12 Cathodic Protection

During the initial Licensing process, a complete survey and tests to determine the need for cathodic protection on Indian Point Unit 2 was made by the A. V. Smith Engineering Company of Narberth, Pennsylvania. Electrical resistivity measurements and a visual inspection of the area away from the river, where the turbine generator building, reactor building, primary auxiliary building and associated facilities are located indicated that the environment is mostly rock with areas of dry sandy clay. The electrical resistivity of the soil ranged from 3,500 to 30,000 ohm-cm with the majority of the readings being above 10,000 ohm-cm. On this basis, it was determined that cathodic protection was not required on underground facilities in areas away from the river or the containment building liner, although a protective coating on pipes was recommended to eliminate any random localized corrosion attack. An analysis of Hudson River water data, obtained from the Con Edison plant chemist, showed the electrical resistivity of the water to vary over an extremely wide range due to salt intrusion from the ocean. The range of resistivity has been from 59 to 10,000 ohm-cm with a large number reading in the 300 ohm–cm area. This value was considered to be extremely corrosive and the following structures in the area near the river were placed under cathodic protection:

1. Circulating water lines.
2. Service water lines.
3. Bearing piles.
4. Sheet piling (earth and water side) and wing wall anchorage system.
5. Metallic structures inside intake structure (traveling screens, bar racks, circulating water pump suction, service water pump suction).

In 2008, a cathodic protection field survey and assessment of underground structures at Indian Point Unit 2 was performed by PCA Engineering of Pompton Lakes, New Jersey. A positive shift in pipe potential was found where the City Water supply piping from the City Water Tank crosses the Algonquin Gas pipes. As a result the City Water supply piping in the vicinity of the gas pipes was placed under cathodic protection.

In 2009, a guided wave assessment of buried piping at Indian Point Unit 2 was performed by Structural Integrity Associates, Inc. of Centennial, Colorado. The assessment identified minor corrosion indications on the Unit 2 CST Condensate supply and return piping in the vicinity of the AFW Pump Building. As a result this piping was placed under cathodic protection.

The cathodic protection system for the Circulating Water lines and the Service Water lines were found not to be functional and the rectifiers were removed. In order to assure the lines will perform their functions, the buried pipes are inspected as part of the Underground Piping and Tank Program. Inspections of buried piping are initially performed using Guided Wave (GW) ultrasonic inspection techniques to locate potential areas of degradation. If significant degradation is detected during the GW inspections, excavation is performed to uncover the affected sections of piping and a direct visual inspection and UT thickness measurements are performed. Repairs and / or replacements are implemented as required to restore degraded piping sections to within the required structural margins of safety.

In addition to the inspections performed as part of the Underground Piping and Tank Program, the nuclear safety related portion of the service water piping is further subjected to pressure and / or flow testing as required by ASME XI, Subsection IWA-5244. Visual inspections on the inside surface of the SW piping are also performed under the GL 89-13, Service Water program. Based on the results of the inspections and testing, the Service Water system is structurally adequate to perform its required safety function.

The cathodic protection system for the Traveling Water Screens and Bar Racks were found not to be effective and the installed cathodic protective systems were retired. The original Traveling Water Screens which were carbon steel were upgraded to stainless steel frames, baskets, and chains. The splash housings are also stainless steel. The Bar Racks, replaced in the mid-1990's are of carbon steel construction and are epoxy coated with a tar epoxy expected to provide corrosion protection over licensed life. The guides for the Screens and Racks are carbon steel channels mounted in a concrete through. The rate of corrosion is slow and the Screens and Racks are on a regular PM cycle that checks for degraded conditions.

The Service Water and Circulating Water pumps suction are not cathodically protected. Rather, the Service Water Pumps suction is inspected and refurbished as part of the Service Water Pumps Preventing Maintenance (PM) activities. The Circulating Water Pumps are inspected and refurbished according to Preventive Maintenance program requirements.

5.1.3.13 Containment - Shear Crack

The arrangement of reinforcing bars in the containment shell is such that a reinforcing bar crosses any potential crack plane. Any cracks resulting from diagonal tension caused by shearing forces will be carried by reinforcing bars, which span across the crack. Thus all shears will be carried by the reinforcing bars and none by the concrete.

The reinforcing bars are almost all continuous throughout the containment structure; however, where a bar terminates this is accomplished by means of a 180 degree hooked bar. In no case are bars simply terminated without providing means for additional anchorage.

Throughout the cylinder, the meridional reinforcing is continuous. Beyond the springline, the bars extend radially toward the center of dome. As the bars reach a 6-in. spacing, which is one-half the required spacing, alternate bars have been dropped off by means of reinforcing splice plates. The splice piece consists of a plate with two Cadweld sleeves welded on the incoming side and one sleeve welded on the outgoing side. Thus, the number of bars present is halved and the spacing is increased to the required 12-in.

This is repeated to the top of the dome where a three layered grid pattern has been used to maintain the continuity of the rebars. The bars in the grid pattern have been Cadwelded to the same type reinforcing splice plates described above, but the Cadweld is beveled to obtain the desired direction of the grid.

At the base in the area of high discontinuity stresses, additional No. 18S bars have been provided. At the point where they were no longer needed, they have been Cadwelded to a No. 11 bar, which is terminated with a 180 degree hook.

All seismic bars have been terminated in a 180 degree hook. In no case was a No. 18S bar terminated in this way since the minimum 180 degree hook could not be provided in a 4-ft 6-in. thick wall.

Radial shear reinforcing stirrups were terminated by hooking around vertical bars.

5.1.4 Containment Penetrations

5.1.4.1 General

In general, a penetration consists of a sleeve embedded in the concrete and welded to the containment liner. The weld to the liner is shrouded by a continuously pressurized channel, which is used to demonstrate the integrity of the penetration-to-liner weld joint. The pipe, electrical conductor cartridge, duct or equipment access hatch passes through the embedded sleeve and the ends of the resulting annulus are closed off, either by welded end plates, bolted flanges or a combination of these. (See Figures 5.1-27 through 5.1-31.)

Differential expansion between a sleeve and one or more hot pipes passing through it is accommodated by using a bellows type expansion joint between the outer end of the sleeve and the outer end plate, as shown on Figure 5.1-30. Pressurizing connections are provided to continuously demonstrate the integrity of the penetration assemblies.

5.1.4.2 Types of Penetrations

5.1.4.2.1 Electrical Penetrations

The electrical penetration system consists of 60 electrical penetrations including the following: 48 Crouse-Hinds, 1 Westinghouse, 10 Conax and 1 spare sleeve (below flood-up level.)

The Crouse-Hinds and Westinghouse types are identical in design (see Fig 5.1-27). This is because Westinghouse took over the Crouse-Hinds manufacturing facility and design after the original plant penetrations were purchased. The design of this type of electrical penetration utilizes a single canister that is sealed at both ends by a combination of metal and ceramic seals. Epoxy layers on both ends provide a physical support for the conductors within the penetration canister. All of the Westinghouse and Crouse-Hinds penetrations are welded to the sleeve inside containment. The entire canister assembly is constantly pressurized by the weld channel pressurization system and monitored for any leakage.

The Conax penetrations are of a modular design consisting of a stainless steel header and 18 independently mounted conductor feedthrough modules (Figure 5.1-28 and 5.1-29), which can be individually removed and relocated. The header plate and the individual feedthrough modules are the pressure-retaining boundary. This type penetration does not have a sealed canister. The conductor modules are threaded into the header plate and the header plate itself is welded to the sleeve, which goes though the containment wall. Leakage monitoring of the Conax penetrations is accomplished by interconnecting ports machined in the header plate to each conductor feedthrough module. A small hole is provided on each conductor feedthrough module stainless steel tubular housing allowing the feedthrough module to be pressurized when the header plate's parts are pressurized. Metal compression fittings (swaging type) are used for mounting the conductor feedthrough modules to the header in a double seal manner. The individual conductors passing through the feedthrough module are surrounded by polysulfone and are sealed (swaged) at each end of the feedthrough housing. The length of the housing (feedthrough tube) is roughly 2-ft longer than the sleeve within, which the penetration is installed. Six of the Conax penetrations are welded to their sleeve outside containment and four are welded to their sleeve inside containment to accommodate differences in the sleeves into which they are welded.

Weld channel rings are used to create a double weld seal between the header plate and the containment sleeve. All of the weld joints necessary maintain containment integrity are monitored for leaks with the weld channel pressurization system.

If a minor leak should develop at any of the plant’s electrical penetrations, a release from inside containment to outside should not occur since each penetration is double sealed and pressurized to maintain a positive pressure (between 49 and 55 psi), which is higher than anticipated containment accident pressures.

5.1.4.2.2 Piping Penetrations

Double barrier piping penetrations are provided for all piping passing through the containment. The pipe is centered in the embedded sleeve, which is welded to the liner. End plates are welded to the pipe at both ends of the sleeve. Several pipes may pass through the same embedded sleeve to minimize the number of penetrations required. In this case, each pipe is welded to both end plates. A connection to the penetration sleeve is provided to allow continuous pressurization of the compartment formed between the piping and the embedded sleeve. . These penetrations are listed as “Hot” in Table 5.2-1. In the case of piping carrying hot fluid, the pipe is insulated and cooling is provided to reduce the concrete temperature adjoining the embedded sleeve. Local areas are allowed to have increased temperatures not to exceed 250ºF. Cooling is provided for hot penetrations through the use of air-to-air heat exchangers. These are made in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, by welding together one flat sheet and one embossed sheet of 10 gauge carbon steel material, the embossment forming coolant passages. The unit is rolled into the form of a cylinder with an outside diameter slightly smaller than the respective inside diameter of the penetration sleeve. The exchanger is placed inside the sleeve and outside the pipe insulation, with the inlet and outlet coolant connections penetrating the sleeve between the outside concrete wall surface and the bellows expansion joint.

The coolant to be used is ambient air fed by a rotary blower, which is backed up with a full sized spare. The isolation features and criteria for piping penetrations are given in Chapter 6. Figure 5.1-30 shows typical hot and cold pipe penetrations.

A total of 107 pipes pass through 53 penetration sleeves, 23 of which are considered thermally hot. In addition, two spare penetration sleeves (capped and pressurized) are available for the possible future addition of piping.

All piping penetrations are designed for normal loads within the stress limits of the ASME Code, Section VIII.

All piping penetrations except main steam and feedwater are designed as anchors for the pipes passing through them and will transmit piping loads to the reinforced concrete wall. The anchorage strength exceeds the maximum combined forces imposed by the effects on the piping penetration of dead load, loads induced from a loss of coolant accident, thermal expansion of the pipe, penetration air pressure, and earthquake loads. The piping penetrations are designed to transmit the above combined loadings to the concrete structure without exceeding the yield strength of penetration steel.

In addition, each piping penetration is designed to withstand, within emergency load criteria, the effect of the rupture of a pipe passing through that penetration at or near the penetration.

The main steam and feedwater penetrations are designed so that the pipes themselves are effectively enclosed for blowdown just inside and just outside the wall. These anchors are designed to prevent a main steam or feedwater pipe rupture from causing a breach of containment at the penetrations. The anchors are designed to 90-percent of yield strength.

All piping penetrating the containment is designed to meet the requirements of USAS B31.1 (1955) Power Piping Code.

Pipes that penetrate the containment building wall and that are subject to machinery-originated vibratory loadings, such as from the reactor coolant pumps, have their supports spaced in such a manner that the natural frequency of the piping system immediately adjacent to the penetrations is greater than the dominant frequencies of the pump. Pipeline vibration was checked during preliminary plant operation and where necessary, vibration dampers were fitted. This checking and fitting effectively eliminates vibrating loads as a design consideration.

5.1.4.2.3 Equipment and Personnel Access Hatches

An equipment hatch has been provided. It is fabricated from welded steel and furnished with a double-gasketed flange and a bolted, dished door. The hatch barrel is embedded in the containment wall and welded to the liner. Provision is made to continuously pressurize the space between the double gaskets of the door flanges, and the weld seam channels at the liner joint, hatch flanges and dished door. Pressure is relieved from the double gasket spaces prior to opening the joints. The personnel hatch is a double door, mechanically latched, welded steel assembly. A quick-acting type equalizing valve connects the personnel hatch with the interior of the containment vessel for the purposes of equalizing pressure in the two systems when entering or leaving the containment. Two spring-loaded check valves in series are installed to allow pressure relief inside the air locks to the containment interior. The 16-ft diameter equipment hatch opening and the 8-ft 6-in. diameter personnel hatch are the only openings, which require special design consideration. The personnel hatch doors are interlocked to prevent both being opened simultaneously and to ensure that one door is completely closed before the opposite door can be opened.

Remote indicating lights and annunciators situated in the control room indicate the door operational status. An emergency lighting and communications system operating from an external emergency supply is provided in the lock interior. Emergency access to either the inner door, from the containment interior, or the outer door, from outside, is possible by the use special door unlatching tools. The design is in accordance with Section VIII of the ASME Code.

The design basis Fuel Handling Accident (FHA) analysis does not credit accident mitigation via Containment isolation subsequent to a postulated fuel assembly drop. However, Containment closure after a FHA event is an option that can reduce total exposure and is a good ALARA practice. The roll-up door serves as a mechanism that will support rapid closure of Containment in the event a radiation release occurs during fuel handling. Containment closure subsequent to a total loss of Residual Heat Removal (RHR) cooling in MODE 6 is accomplished through installation of the equipment hatch or temporary closure plate. The roll-up door is currently not considered a suitable device that can be credited for Containment closure subsequent to a loss of RHR, pending NRC approval of an outstanding license Amendment request.

5.1.4.2.4 Special Penetrations

1. Fuel Transfer Penetration - A fuel transfer penetration is provided for fuel movement between the refueling transfer canal in the reactor containment and the spent fuel pit. The penetration consists of a 20-in. stainless steel pipe installed inside a 24-in. pipe. The inner pipe acts as the transfer tube and is fitted with a pressurized double-gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pit. This arrangement prevents leakage through the transfer tube in the event of an accident. The outer pipe is welded to the containment liner and provision is made by use of a special seal ring for pressurizing all welds essential to the integrity of the penetration during plant operation. Bellows expansion joints are provided on the pipes to compensate for any differential movement between the two pipes or other structures. Figure 5.1-31 shows a sketch of the fuel transfer tube.

2. Containment Supply and Exhaust Purge Ducts - The ventilation system purge ducts are each equipped with two quick-acting, tight-sealing valves (one inside and one outside of the containment) to be used for isolation purposes. The valves are manually opened for containment purging, but are automatically closed upon receipt of a safety injection signal or high-containment radiation signal. The space between the valves is pressurized above design pressure while the valves are normally closed during plant operation. See Section 5.3,

Containment Ventilation System, and Section 6.4, Containment Air Recirculation Cooling System.

Seismic Class I debris screens inside the primary containment protect the primary containment isolation valves in the containment purge and pressure relief exhaust ducts from debris that may inhibit their correct operation. The screens are stainless-steel wire mesh and are mounted over the exhaust ducts.

Two solenoid-controlled, pneumatically operated butterfly valves are provided for each purge penetration, one on each side of the containment building wall. Two penetrations, one supply and one exhaust, are required. Valves are spring-loaded to fail closed.

The space between the valves is pressurized from the pressurization system through an electrically operated three-way solenoid valve. This pressure is maintained only when valves are closed and must be relieved before butterfly valves can be opened. Failure to release this pressure will prevent the inside containment valves from opening. By procedure the outside containment valves are opened after the inside containment valves are open.

Failure of any of the valves to open will prevent the containment building purge supply fan from running. Tripping of the containment building purge supply fan will automatically close the inside containment butterfly valves. By procedure the outside containment butterfly valves must then be closed. When these valves are closed the space between the valves is automatically pressurized. Failure of any valves to close will prevent the adjacent space from being pressurized and will sound the loss of pressurization alarm. Loss of pressure for either zone will be displayed by individual indicating lights at the main control board.

The valve control solenoids for the inside containment isolation valves FCV-1170 and FCV-1172 and pressurization solenoids are controlled from a single control switch on the fan room control panel. The valve control solenoids for outside containment isolation valves FCV-1171 and FCV-1173 are controlled from a switch in the control room. The cycle is initiated by setting the fan room control switch to the "open" position. This will energize the pressurization alarm.

When the pressure between the valves has been relieved, the valve control solenoids for the inside containment isolation valves are energized and these two valves are opened. If for any reason, either of the two inside containment isolation valves fail to open within a given time after the cycle is initiated, both of these valves will close and pressure will be restored. The circuit is interlocked to prevent inadvertent opening of the valves during a safety injection condition.

Once the inside containment purge valves have been opened, the operator has a predetermined time to place the control switch for the outside containment purge valves to the "open" position and once opened to start the purge supply fan. Failure to do so will cause the inside containment purge valves to close.

Position indicating lights for each of the four valves are provided on the fan room control panel and the main control board.

3. Sump Penetrations - The piping penetration in the containment sump area is not of the typical sleeve-to-liner design. In this case, the pipe is welded directly to the base liner. The weld to the liner is shrouded by a test channel, which is used to demonstrate the integrity of the liner.

5.1.4.3 Design of Containment Penetrations

5.1.4.3.1 Criteria

The liner is basically not a load-carrying member. Because it is subjected to strains imposed by the reinforced concrete, the liner has been reinforced at each penetration in accordance with the ASME Code Section VIII. The weldments of liner to penetration sleeve are of sufficient strength to accommodate stress concentrations and adhere strictly to ASME Code Section VIII requirements for both type and strength. The penetration sleeves and plates are designed to accommodate all loads imposed on them under operating conditions (thermal effects and internal penetrations and test pressures) and accident conditions (loads resulting from all strains, internal pressures, and seismic movements). All reinforcing bars except stirrups and facing bars that are not counted on to carry any load are continuous around the openings.

Liner stress is imposed on the cylindrical penetration as a circular uniform load acting around the circumference of the penetration. The liner plate is locally thickened at the penetrations to take care of additional stresses.

5.1.4.3.2 Materials

The materials for penetrations, including the personnel and equipment access hatches, together with the mechanical and electrical penetrations, are carbon steel, conforming with the requirements of the ASME Nuclear Vessels Code and exhibiting ductility and welding characteristics compatible with the main liner material. As required by the Nuclear Vessels Code, the penetration materials meet the necessary Charpy V-notch impact values at a temperature 30°F below lowest service metal temperature, which is 50°F within containment and -5°F outside the containment.

The stainless steel bellows of the hot penetration expansion joints were protected from damage in transit and during construction by sheet metal covers fastened in place at the fabricator's shop.

1. Piping Penetrations: Materials

Piping Penetration Material	Specification
Penetration Sleeve - 12-in. dia. and under	ASTM-A333, Gr. 1
Over 12-in. dia.	ASTM-A201, Gr. B
(see exception below)	normalized to A300 CL. 1, Firebox
- 22-in. dia. containment	ASTM-A53, Gr. B sump suction
- Rolled shapes	ASTM-A36, A131, Gr. C

2. Electrical Penetrations: Materials

The penetration sleeves to accommodate the electrical penetration assembly cartridges are schedule 80 carbon steel in accordance with ASTM-A333, Gr. 1, except where otherwise noted. The electrical cartridges have been secured to the penetration sleeve so that all possible leak paths between the cartridge and sleeve will be blocked by a pressurized zone.

3. Access Penetrations: Materials

The equipment and personnel access hatch material is as follows:

Item	Material Specification
Equipment hatch insert	ASTM A516, Gr. 60 normalized to ASTM A300, CL. 1, Firebox
Equipment hatch flanges	ASTM A516, Gr. 60 normalized to ASTM A300, CL. 1, Firebox
Equipment hatch head	ASTM A516, Gr. 60 normalized to ASTM A300, CL. 1, Firebox
Personnel hatch	ASTM A516, Gr. 60 normalized to ASTM A300, CL. 1, Firebox

5.1.4.4 Leak Testing of Penetration Assemblies

A preoperational proof test was applied to each penetration by pressurizing the necessary areas to 54 psig. This pressure was maintained for a sufficient time to allow soap bubble and Freon sniff tests of all welds and mating surfaces. Any leaks found were repaired and retested; this procedure was repeated until no leaks existed.

5.1.4.5 Construction

The qualification of welding procedures and welders has been in accordance with Section IX, "Welding Qualifications," of the ASME Boiler and Pressure Vessel Code. The repair of defective welds has been in accordance with Paragraph UW-38 of Section VIII, "Unfired Pressure Vessels."

5.1.4.6 Testability of Penetrations and Weld Seams

All penetrations, the personnel air lock, and the equipment hatches are designed with double seals, which will be normally pressurized at or above the containment design pressure. Individual testing at 115-percent of containment design pressure is also possible.

The containment ventilation purge ducts are equipped with double isolation valves and the space between the valves is permanently piped into the penetration pressurization system. The space can be pressurized to 115-percent of design pressure when the isolation valves are closed. The purge valves fail in the closed position upon loss of power (electric or air).

All welded joints in the liner have steel channels welded over them on the inside of the vessel. During construction, the channel welds were tested by means of pressurizing sections with Freon gas and checking for leaks by means of a Freon sniffer. These welds were also then continuously pressurized at 50 psig.

5.1.4.7 Accessibility Criteria

Limited access to the containment through personnel air locks is possible with the reactor at power or with the primary system at design pressure and temperature at hot shutdown. After shutdown, the containment vessel is purged to reduce the concentration of radioactive gases and airborne particulates. This purge system has been designed to reduce the radioactivity level to doses defined by 10 CFR Part 20 for a 40-hr occupational work week within 2 to 6 hr after plant shutdown. Since negligible fuel defects are expected for this reactor, much less than the 1-percent fuel rod defects used for design and purging of the containment is normally accomplished in less than 2 hr. To ensure removal of particulate matter and radioactive gases, the purge air is passed through a high efficiency and charcoal filters before being released to the atmosphere through the purge vent. The primary reactor shield has been designed so that access to the primary equipment is limited by the activity of the primary system equipment and not the reactor.

5.1.4.8 Penetration Design – Computations

The penetration sleeves and end plates are designed to accommodate all loads imposed on them. The sleeve and end plate loads include the effects of internal pressure; concentrated loads imposed by the sleeve anchors on the concrete as the anchors strain in conjunction with wall movement under both operating and accident conditions; thermal effects due to both gradient and thermal reactions of the particular item passing through the sleeve; shear, bending, and compression due to accident end pressures; and shear and bending due to seismic movements of the particular item passing through the penetration. The sleeve and expansion joint are designed to remain within ASME Code Section VIII stress limitations with small strains under all or any combinations of loadings mentioned above.

For design computations of penetrations and the shell adjacent to them, see Figures 5.1-32 and 5.1-33. In Section 5.1.4.8.1, the formula for radial deformation of a hole in a plate subjected to biaxial stresses is determined by performing an integration of the tangential strains around the periphery of the hole.

In Section 5.1.4.8.2, the relationship between the deflection determined from above to the final plate and penetration sleeve deformations is developed and the formulas for stress in the liner and the stress in the penetration sleeve are developed.

Section 5.1.4.8.3 shows a summary of the liner and penetration stresses and states the assumptions made in the analysis.

In addition, thermal loads have been investigated for their effect on the shell adjacent to the penetration sleeve and found to be insignificant (38 psi bearing stress on the concrete is the maximum stress on the concrete shell).

5.1.4.8.1 Radial Deformation of a Hole in a Plate

From Reference 17, page 81

$$\sigma o = S - 2\,S\cos 2\theta + [S' - 2S'\cos(2\theta - \pi)]$$

where

S = Horizontal stress in liner
S' = Vertical stress in liner

$$\delta D = \frac{1}{E}\int_{o}^{\pi}\left(S - 2S\cos 2\theta + \left[S' - 2S'\cos\left(2\theta - \pi\right)\right]\right) r\sin\theta d\theta$$

$$\delta D = \frac{r}{E}\left[\int_{o}^{\pi} S\sin\theta d\theta - 2S\int_{o}^{\pi}\cos 2\theta\sin\theta d\theta + S'\int_{o}^{\pi}\sin\theta d\theta - 2S'\int_{o}^{\pi}\cos\left(2\theta - \pi\right)\sin\theta d\theta\right]$$

$$\int\cos\left(2\theta - \pi\right)\sin\theta d\theta = -\int\cos 2\theta\sin\theta d\theta$$

$$= -\int\left(1 - 2\sin^2\theta\right)\left(\sin\theta\right)d\theta$$

$$= -\int\left(\sin\theta - 2\sin^3\theta\right)d\theta$$

$$= -\left[\left(-\cos\theta\right) - 2\frac{\sin^2\theta\cos\theta}{3} + \frac{2}{3}\int\sin\theta d\theta)\right]$$

$$= -\left(-\cos\theta + 2/3\sin^2\theta\cos\theta + \frac{4}{3}\cos\theta\right)$$

therefore

$$\int\cos(2\theta - \pi)\sin\theta d\theta = \frac{-\cos\theta}{3} - \frac{2}{3}\sin^2\theta\cos\theta$$

$$\delta = \frac{r}{E}\left[-S\cos\theta - 2S\left(\frac{\cos\theta}{3} + \frac{2}{3}\sin^2\theta\cos\theta\right) - S'\cos\theta - 2S'\left(\frac{-\cos\theta}{3} - \frac{2}{3}\sin^2\theta\cos\theta\right)\right]_{o}^{\pi}$$

$$\delta = \frac{r}{E}\left[\left(S + \frac{2}{3}S + S' - \frac{2}{3}S'\right) - \left(-S - \frac{2}{3}S - S' + \frac{2}{3}S'\right)\right]$$

$$\delta = \frac{r}{E}\left[2S + \frac{4}{3}S + 2S' - \frac{4}{3}S'\right]$$

$$\delta = \frac{r}{E}\left[\frac{10}{3}S + \frac{2}{3}S'\right]$$

$$\delta = \frac{2}{3}\frac{r}{E}\left[5S + S'\right]$$ (for stresses in the same direction)

$$\delta = \frac{2}{3}\frac{r}{E}\left[5S - S'\right]$$ (for stresses in the opposite direction)

5.1.4.8.2 Plate and Sleeve Deformation

$$\Delta_{UN} = \Delta_{Pl\,Res.} + \Delta_{Sleeve}$$

$$\Delta_{UN} = \frac{S_1}{E}(1-\nu)R + \frac{S_l\left(t_{pl}\right)R^2\lambda}{2Et_{sleeve}}^{*}$$

$$\Delta_{UN} = \frac{S_1}{E}\left[R(1-\nu) + \frac{t_{pl}\;\;R^2\lambda}{2t_{sleeve}}\right]$$

$$S_1 = \frac{\Delta_{UN}E}{R\left[(1-\nu) + \frac{t_{pl}\;R\lambda}{2t_{sleeve}}\right]}$$

$$S_{sleeve} = \frac{S_1\left(t_{pl}\right)R\lambda}{2t_{sleeve}}$$

$$S_{sleeve} = \frac{\Delta_{UN}E\;\;t_{p1}\;\;R\lambda}{R\left[(1-\nu) + \frac{t_{p1}\;R\lambda}{2t_{sleeve}}\right]2t_{sleeve}}$$

where:

$$\lambda = \left[\frac{3\,(1-\nu^2)}{R^2t^2_{sleeve}}\right]^{\frac{1}{4}}$$

*S_1 = Stress in Liner
t_{pl} = plate thickness, in.

R = radius, in.
ν = poisson ratio
E = modulus of elasticity

5.1.4.8.3 Summary

Penetration	Stress in Sleeve (ksi)	Stress in Liner (ksi)
Air purge	-23.8	-19.5
Main steam	-33.4	-27.94
Typical mech. Penetration	-31.0	-31.1
Electrical penetration		
A) C and T[1]	-22.5	-29.5
B) T and T[1]	+18.2	+19.7
Fuel transfer		
A) C and T[1]	-25.7	-25.6
B) T and T[1]	+20.8	+16.6

[*Note – 1. First letter represents direction of vertical liner stress; second letter represents direction of horizontal liner stress, C signifies compression and T signifies tension.*]

A) Ignores effects of insulation in the vertical direction.

B) Considers effects of insulation.

Conservative Assumptions

1. The weld pressurization channel stiffens the area.

2. The liner alone was designed for stress concentration effects while the cracked concrete was ignored.

3. The unrestrained growth is based on maximum growth from a stress concentration consideration.

4. The main steam and mechanical penetrations have been considered in a noninsulated zone when they are just inside the insulated zone. The compression in the hoop direction will be greatly reduced or perhaps go into tension, thus reducing the stresses.

5. The allowable stress in the sleeve is 56,700 psi except for the stainless steel fuel transfer penetration, which is 49,500 psi. These values come from Table N-421 and Figure N-414 of the ASME Nuclear Vessel Code Section III.

5.1.5 <u>Primary System Supports</u>

In 1989, the NRC approved changes to the design bases with respect to dynamic effects of postulated primary loop pipe ruptures, as discussed in Section 4.1.2.4.

In 2000, an analysis (Reference 19) of the reactor coolant loop and its component supports, which incorporates the NRC approved changes, was performed to reflect the replacement steam generator and removal of sixteen of the original twenty-four steam generator support frame hydraulic snubbers. The analysis also reflected the de-activation of the original horizontal and vertical pipe rupture restraints, located on the cross over legs at the steam generator end. Based on this revised analysis, it was concluded that the Unit 2 reactor coolant system can withstand the combination of blowdown and seismic loads within acceptable stress limits. By reducing the number of snubber and de-activating the rupture restraints the extent of maintenance, inspection and testing requirements is reduced and the reliability of the Reactor Coolant System is enhanced by reducing the possibility of equipment malfunction. In 2003, the reactor coolant loop and its component supports were re-analzed due to a power uprate. This latest analysis does not consider the coincident combination of blowdown and seismic loads.

The original design basis is described in the following paragraphs.

The primary system supports, steam generator, reactor coolant pump, pressurizer, and reactor vessel were designed to withstand pipe break or seismic acceleration based on the following:

1. The break is either a circumferential or longitudinal pipe rupture of area equivalent to the pipe cross section occurring anywhere in the system piping. The longitudinal rupture occurs at any point 360 degrees around the pipe. The support system is designed to withstand the steady thrust equivalent to the product of system operating pressure and pipe rupture area without exceeding yield stress in the support members. The stress limits on the vessels and piping are tabulated in Section 1.11. The component supports prevent rupture of reactor coolant piping in the remaining intact loops which could result from an assumed rupture in any one loop, thereby ensuring that the path for safety injection flow to the core is available. Additionally, the supports are designed to prevent secondary piping rupture as a result of rupture in the primary loop and vice versa.

2. The nuclear steam supply system and its support system are designed such that the nuclear steam supply system is capable of continued safe operation for the combination of normal loads and the design earthquake loading. The equipment and supports operate within normal design limits for the design earthquake. The system and its supports are also designed to withstand the maximum potential earthquake without loss of function. The seismic response curves for both the design and maximum potential earthquake and the stress limits are presented in Section 1.11. Component loads are obtained from the curve using the appropriate period and damping.

3. The primary system supports were not originally designed to resist combined seismic and accident loads. They were designed as statically uncoupled component supports.

   A complete reactor coolant system loop, including the steam generator and the reactor coolant pump supports, has been analyzed for combined dead, seismic, and blowdown loads. Stresses were determined by means of the three-dimensional frame computer program, STRUDL. The dead load assumed is the flooded weight of the component. The seismic load considered is 0.6g horizontal acceleration times the flooded mass of the component at the center of gravity of the component acting in the N-S, E-W, NW-SE and SW-NE directions analyzed separately. The horizontal earthquake component acting on the steam generator is assumed to be carried by the upper steam generator. The vertical component of earthquake is assumed equal to 0.4g acting simultaneously with the horizontal load at the center of gravity of both the pump and steam generator. The system was analyzed for each separate accident or pipe rupture resulting in a jet load equal to 1500 kips as shown in Figure 5.1-34.

The combined dead plus seismic plus accident maximum resultant member axial stress and axial plus bending stress (in parentheses) for the steam generator and pump supports is shown in Figures 5.1-35 through 5.1-42 (stresses are expressed in ksi.) The section views of the support shown can be identified by the isometric views of the pump and steam generator supports shown in Figures 5.1-43 and 5.1-44. Negative values indicate compression and positive values indicate tensile stress. Since response of the primary systems is elastic, deformations are very small and were not considered as design parameters required to verify the design adequacy of the supports.

It should be noted the stresses shown are not for a particular combined blowdown or seismic load case but rather the worst combination for a given member; hence, the values shown are upper limits for each member and could not in fact actually occur in the combination shown. It should also be noted that the primary support structures are designed as trusses rather than frames hence the bending stresses indicated are secondary in nature.

5.1.5.1 Steam Generator

The steam generators are supported within a caged structural system, consisting of four connected trusses, all welded together, fabricated of carbon steel members, with provisions for limited movement of the structure in a horizontal direction to accommodate piping expansion with a system of "Lubrite" plates, hydraulic snubbers, guides, and stops. The "Lubrite" plates, hydraulic snubbers, guides, and stops were originally designed as a rigid support to resist the action of seismic and pipe break loads. In 2000, the number of hydraulic snubbers supporting the steam generator frame in the direction of the hot leg, has been reduced from the original six down to two per steam generator. The two remaining snubbers are located at the upper support point of the frame at Elevation 92'-0". The analysis of the reactor coolant loop and of the steam generator support structure accounts for the replacement steam generator and for the reduced number of hydraulic snubbers. The following are loading conditions that the structure was originally designed to resist:

1. Vertical dead weight of pipe and vessel, flooded = 1,000 kips

2. Seismic loads:

   a. Horizontal load of 474 kips acting at the centroid of the steam generator vessel, located near the top of the support structure, which is directly transferred to the hydraulic snubbers, guides, and stops, and in turn to the bottom of the 2-ft thick concrete operating floor slab at elevation 93-ft.

   b. Vertical load of 320 kips transferred as axial load to the base plates and anchor bolts at elevation 46-ft.

3. Primary system - longitudinal pipe rupture:

   a. Reaction at the nozzle of the steam generator from the pipe between the reactor and the steam generator elbow, produces a force of 1090 kips in any direction and an overturning moment or torsional moment of (1090 kips x 4.25-ft) 4632-ft-kips. Overturning and torsional moments are resisted by the support system at elevation 46-ft and horizontal forces are distributed, through the truss action, to elevations 46-ft and 93-ft.

   b. Reactions at the nozzle of the steam generator from the pipe between the steam generator elbow and reactor coolant pump elbow, produces a force of 850 kips in any direction and a torsional moment or overturning moment of (850 kips x 5.0-ft) 4250-ft-kips. Overturning and torsional moments are resisted by the support system at elevation 46-ft, and horizontal forces are distributed, through the truss action, to elevations 46-ft and 93-ft.

4 Primary system - circumferential break:

   a. Reactions at the nozzle of the steam generator from the pipe between the reactor and steam generator produces a horizontal force of 1490 kips. This force is transferred through the vessel support to the two vertical trusses of the structural system, which in turn, transmits it as horizontal reactions at the slabs at elevations 46-ft and 93-ft. The moment produced by this force is (1490 kips x 2-ft) 2980-ft-kips and is less than the dead load resisting moment (500 kips x 10-ft) 5000-ft-kips, and the vertical forces at elevation 46-ft are all compressive, no uplift.

   b. Reactions at the nozzle of the steam generator from a pipe between the steam generator and the reactor coolant pump produces a horizontal force of 1700 kips plus an overturning moment of (1700 kips x 4.25-ft) 7225-ft-kips, or a vertical force of 1700 kips and an overturning moment of (1700 kips x 5.33-ft) 9061-ft-kips. The horizontal force and moments are transferred to the structural system and the reactions are resisted at the slabs at elevations 46-ft and 93-ft, or the vertical force and moment are resisted at elevation 46 ft.

5. Secondary system - longitudinal rupture in steam pipe:
   Reactions at the nozzle of the steam generator from the steam pipe longitudinal rupture at the top of the vessel produce:

   a. Horizontal force of 600 kips and a torsional moment of 2400-ft-kips. Horizontal force is transferred through the vessel to the structural support system, which in turn transmits it as horizontal reactions to the slabs at elevations 46-ft and 93-ft. The torsional moment is transferred through the vessel to the structural system, which in turn, transmits it to the base at elevation 46-ft.

   b. Vertical upward or downward force of 600 kips and an overturning moment of 2400-ft-kips. Upward forces are overcome by the operating weight of the steam generator. Downward force is added to the operating weight and transferred to the base at elevation 46-ft. Overturning moment is transferred through the vessel supports to the structural system, which in turn, transmits it as vertical reactions at the base, elevation 46-ft.

6. Secondary system - circumferential break in steam pipe:

   Reaction at the nozzle of the steam generator from the steam pipe guillotine break at the top of the vessel produces a horizontal force of 600 kips. This force is transferred through the vessel to the structural system, which in turn transmits it as horizontal reactions of 1085 kips at elevation 93-ft and 485 kips at elevation 46-ft.

7. Secondary system - feedwater pipe breaks:

   The reactions from circumferential and longitudinal pipe breaks in the feedwater system are resisted in a manner similar to steam pipe breaks listed under preceding sections (5) and (6), but are much smaller in magnitude. Maximum longitudinal 1600-ft-kips, maximum circumferential 200 kips.

5.1.5.2 Reactor Coolant Pump

The reactor coolant pump is supported on a three-legged structural system consisting of three connected trusses fabricated of carbon steel members, structural sections and pipe, supported from elevation 48-ft-6-in. Provisions for limited movement of the structure in any horizontal direction to accommodate piping expansion is accomplished with a sliding "Lubrite" base plate arrangement and a system of tie rods and anchor bolts that restrain the structure from movement beyond the calculated limits. To improve the ability of the reactor coolant pumps to meet combined LOCA and seismic loads, two of the reactor coolant pump holddown bolts have been replaced with higher strength ASTM A540 steel bolts.

The following are loading conditions that the structure was originally designed to resist:

1. Vertical dead weight of pipe and pump flooded = 206 kips.

2. Seismic:
   a. Horizontal load of approximately 117 kips acting at the centroid of the pump assembly, which is transferred by the structural system and piping to the tie rods and base of the supporting structure at elevation 48-ft-6-in. This load includes the seismic effect of the support self-weight.

   b. Vertical seismic load of approximately 78 kips transferred directly as axial load to the base plates and anchor bolts. This load includes the seismic effect of the support self-weight.

3. Primary system - longitudinal rupture:
   a. Reaction at the nozzle of the pump from a pipe break in the pipe between the steam generator elbow and pump elbow produces a torsional moment of 3825-ft-kips, together with a horizontal force of 850 kips or an overturning moment of 3825-ft-kips, together with a vertical up or down force of 850 kips. Torsional forces are resisted by the structural stability of the primary piping connected to the pump.

Reactions from horizontal forces are resisted by the tie rods connected to the steam generator and reactor support structures. Forces caused by an overturning moment are resolved into horizontal and vertical components, which are resisted by tension in the anchor bolts, axial load on the foundations, and tension in the tie rods.

b. Reaction at the nozzle of the pump from a pipe break in the pipe between the pump and the reactor, produces a torsional moment of 6880-ft-kips, together with a horizontal force of 1165 kips, or an overturning moment of 6880-ft-kips, together with a vertical up or down force of 1165 kips.

Torsional forces are resisted by the structural stability of the primary piping connected to the pump. Reactions from the horizontal forces are resisted by the tie rods connected to the walls.

Forces caused by an overturning moment are resolved into horizontal and vertical components, which are resisted by:

(1) Tension in the anchor bolts.
(2) Axial load on the foundations.
(3) Tension in the tie rods.

4. Primary system - circumferential break:

a. Reactions at the nozzle of the pump from a pipe break in the pipe between the steam generator and pump produces a horizontal force on the structure of 1700 kips. This force is resisted directly by the bumper located against the elbow of the pipe. Components of the force are then transferred to the base of the structure and the tie rods connecting the pump support to the steam generator support system.

b. Reactions at the nozzle of the pump from a pipe break in the pipe between the pump and the reactor produces a torsional moment of 3240-ft-kips and a horizontal force of 1340 kips on the structure.

Torsional forces are resisted by the structural stability of the remaining primary piping connected to the pump.

Reactions from the horizontal forces are resisted by tie rods connected to the walls.

5.1.5.3 Pressurizer

Pressurizer is supported on a free-standing structural system, consisting of six connected trusses fabricated of carbon steel members, all welded together and secured at the base by anchor bolts at elevation 46-ft.

The following are loading conditions that the structure has been designed to resist:

1. Vertical dead weight of pipe and vessel flooded is 360 kips. The self-weight of the support is 21 kips.

2. Seismic:

   a. Horizontal seismic load of 174 kips acting at the centroid of the pressurizer vessel, which coincides in elevation with the slab at elevation 95-ft, is directly transferred through the concrete embedded guides to the slab. This load excludes the seismic effect of the support self-weight.

   b. Vertical seismic load of 123 kips transferred through the structural system as axial forces to the base plates and anchor bolts at elevation 46-ft. This load excludes the seismic effect of the support self-weight.

3. Longitudinal pipe rupture

   a. Reaction at the surge pipe nozzle of the pressurizer produces either a torsional moment of 734-ft-kips and a horizontal force of 234 kips or an overturning moment of 734-ft-kips and a horizontal or vertical force of 234 kips.

      These moments and forces are resisted by the structural system and transferred to the base at elevation 46-ft.

4. Circumferential pipe break:

   a. Reaction at the surge pipe nozzle of the pressurizer produces a horizontal force of 234 kips and an overturning moment of 734-ft-kips.

      These moments and forces are resisted by the structural system and transferred to the base at elevation 46-ft.

5.1.5.4 Reactor Vessel Support Girder

The reactor vessel is supported on four cooling plates that are fastened to the top flange of a circular box section ring girder, fabricated of carbon steel plates. The bottom flange of the girder is in continuous contact with a nonyielding concrete foundation.

In addition to the reactor vessel weight and piping reactions of the girder has been designed to support the conditions of loading for pipe break and seismic forces as outlined in Figure 5.1-45.

5.1.5.5 Reactor Vessel Rupture

The reactor pressure vessel is enclosed by a 6-ft thick circular reinforced concrete shield wall that is designed to sustain the internal pressure and provide missile protection for the containment liner in the highly unlikely failure of the reactor vessel due to a longitudinal split. All stresses will be maintained within specified minimum ultimate rebar tensile stress.

In the event of a circumferential reactor break, the 0.25-in. basemat liner plate at the bottom of the containment reactor cavity pit directly under the reactor vessel is protected by 2-ft of concrete with a 1-in. steel liner plate embedded on top of the concrete. Directly below the reactor cavity pit containment basemat liner plate, 4.5-ft of concrete is poured on rock. Refer to Figures 5.1-46 through 5.1-51.

As discussed in Section 5.1.3.7, in the event of reactor vessel failure, a pressure build up of 1000 psi and rebar stresses of 86 ksi (assuming all concrete is cracked) inside the pit due to release of reactor contents is assumed. Since the integrity of the wall is not jeopardized, the integrity of the vessel support that is supported on the wall will not be jeopardized. Deflection of the shield wall will not cause large stresses in the vessel support since a lubricated surface is provided on the shoes, allowing the vessel support to slide.

5.1.5.6 Circumferential Cracking

The worst circumferential crack location from the standpoint of downward missiles is just below the reactor coolant system piping nozzles. As the following calculations show, this missile will not violate the containment structure and liner integrity.

As a consequence of this circumferential crack, the downward missile represented by bottom vessel head has the following characteristics at the time of impact on the cavity floor:

1. Weight: 381,000 lb
2. Cross sectional area of crater: 63-ft^2
3. Downward velocity: 213-ft/sec
4. Concrete crushing strength: 4000 psi

The depth of penetration has been calculated by using the Petri formula for penetration into an infinite, thick concrete slab, as reported in Nav. Docket P-51:

$$D = K\frac{W}{A} \log_{10} \left(1 + \frac{V^2}{215{,}000}\right)$$

where:

D = depth of penetration, ft
K = penetration coefficient for 4000 psi concrete
W = missile weight, lb
A = missile area, ft^2
V = missile velocity, ft/sec

The following parameters have been used:

$K = 2.8 \times 10^{-3}$
W = 381,000 lb
A = 63-ft^2
V = 213 ft/sec

The result is a depth of penetration of 1.4-ft.

As mentioned above, the 0.25-in. basemat liner is covered by 2-ft of concrete with a 1-in. steel plate on top. As it can be readily seen, even neglecting the 1-in. steel plate in the penetration calculations, the containment liner will not be reached.

5.1.5.7 Longitudinal Splitting

The cavity wall is designed to withstand the forces and internal pressurization associated with a longitudinal split without gross damage. See Section 5.1.3.7 for a discussion of the analysis of this assumed accident condition.

5.1.6 Containment Structure Design Evaluation

5.1.6.1 Reliance On Interconnected Systems

The containment leakage limiting boundary is provided in the form of a single, carbon steel liner on the vessel having double barrier weld channels and penetrations. Each system whose piping penetrates this boundary is designed to maintain isolation of the containment from the outside environment. Provision is made to continuously pressurize penetrations and weld channels and to monitor leakage from this pressurization.

5.1.6.2 System Integrity and Safety Factors

Pipe Rupture - Penetration Integrity - The penetrations for the blowdown and sample lines are designed so that the penetrations are stronger than the piping system and so that the vapor barrier will not be breached due to a hypothesized pipe rupture. The pipe rupture loads for the main steam and feedwater lines are resisted by the supports located away from their penetrations and do not affect the integrity of the penetrations for these lines.

Major Component Support Structures - The support structures for the major components are designed to resist all thrust forces, moments and torques associated with either a reactor coolant system or main steam pipe break. All primary structural steel elements are designed for stresses not exceeding yield stress due to these forces.

Containment Structure Components Analyses - The details of radial, longitudinal, and horizontal shear analyses for the containment reinforced concrete are given in Section 5.1.3.

5.1.6.3 Performance Capability Margin

The containment structure is designed based upon limiting load factors, which are used as the ratio by which accident and earthquake loads are multiplied for design purposes to ensure that the load/deformation behavior of the structure is one of elastic, low strain behavior. This approach places minimum emphasis on fixed gravity loads and maximum emphasis on accident and earthquake loads. Because of the refinement of the analysis and the restrictions on construction procedures, the load factors primarily provide for a safety margin on the load assumptions. Load combinations and load factors used in the design, which provide an estimate of the margin with respect to all loads, are tabulated in Section 5.1.2.

5.1.7 Liner Insulation

Insulation is provided on approximately the first 43-ft of the containment liner to limit the temperature rise in the liner under accident conditions to 80°F above ambient and thereby avoid excessive liner compressive stress during the accident. The first 18-ft (elev. 46-ft to 64-ft, except in the piping penetration area in the southeast quadrant where the insulation rose only 16-ft to the 62-ft elevation) was installed as part of the original containment design. In 1973 an additional 25-ft (elev. 64-ft to 89-ft) was added. The first 18-ft (elev. 64-ft to 82-ft) covers the entire circumference of the liner while the upper 7-ft (elev. 82-ft to 89-ft) only covers part of the circumferential area in the north and south-southwest quadrants where the main steam and feedwater lines extend up along the crane wall. The insulation panels are attached to the steel containment liner by means of 3/16-in. diameter stainless steel studs welded to the liner on the basis of six per panel. The insulation panels are protected by stainless steel jacketing on the exposed faces and sealed at the joints. Details of the insulation installation are given in Table 5.1-2.

The insulation has been designed to meet the following operational requirements:

1. Normal operating temperature of 120°F. (The maximum normal operating temperature of the containment was changed from 120°F to 130°F by Amendment 149 to the Facility Operating License DPR-26 for IP-2 dated March 27, 1990. Evaluations performed show the insulation material used on the containment liner is adequate for use at the higher operating temperature.)

2. Under accident conditions, the rise in liner temperature not to exceed 80°F. The analyses performed to support the Stretch Power Uprate (SPU) also performed analyses of the containment liner under the most limiting conditions for liner stress and showed a temperature rise well under allowed 80°F.

3. Insulation panels to be rated non-burning in accordance with ASTM procedure D−1692.

4. To be removable by sections for inspection of the containment liner.

5.1.8 Minimum Operating Conditions (For Containment Integrity)

Containment integrity internal pressure limitations and leakage rate requirements are established in the facility Technical Specifications.

5.1.9 Containment Structure-Inspection And Testing

5.1.9.1 Initial Containment Leakage Rate Testing

Criterion: Containment shall be designed so that integrated leakage rate testing can be conducted at the peak pressure calculated to result from the design basis accident after completion and installation of all penetrations and the leakage rate shall be measured over a sufficient period of time to verify its conformance with required performance. (GDC 54)

After completion of the containment structure and installation of all penetrations and weld channels, an initial integrated leakage rate test was conducted at the containment design pressure (47 psig), maintained for a minimum of 24 hr, verifying that the leakage rate is no greater than 0.1-percent by weight of the containment volume per day at design basis accident conditions. This leakage rate test was performed using the absolute method. In addition, a reduced pressure integrated leakage rate test was conducted at a pressure not less than 50 percent of the containment design pressure and maintained for a minimum of 24 hr.

5.1.9.2 Periodic Containment Leakage Rate Testing

Criterion: The containment shall be designed so that an integrated leakage rate can be periodically determined by test during plant lifetime. (GDC 55)

The containment is tested in accordance with 10 CFR 50 Appendix J as discussed in section 5.1.12.

A leak rate test at the containment design pressure using the same method as the initial leak rate test can be performed at any time during the operational life of the plant, provided the plant is not in operation and precautions are taken to protect instruments and equipment from damage.

5.1.9.3 Provisions for Testing of Penetrations

Criterion: Provisions shall be made to the extent practical for periodically testing penetrations, which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at the peak pressure calculated to result from occurrence of the design basis accident. (GDC 56)

Penetrations are designed with double seals, which are continuously pressurized above accident pressure. The large access openings such as the equipment hatch and personnel air locks are equipped with double gasketed doors and flanges with the space between the gaskets connected to the pressurization system. The system uses a supply of clean, dry, compressed air that will place the penetrations under an internal pressure above the peak calculated accident pressure.

A permanently piped monitoring system is provided to continuously measure leakage from all penetrations.

Leakage from the monitoring system is checked by continuous measurement of the integrated makeup air flow. In the event excessive leakage is discovered, each penetration can then be checked separately at any time.

5.1.9.4 Provisions for Testing of Isolation Valves

Criterion: Capability shall be provided to the extent practical for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits. (GDC 57)

Capability is provided to the extent practical for testing the functional operability of valves and associated apparatus during periods of reactor shutdown.

Initiation of containment isolation employs coincidence circuit, which allow checking of the operability and calibration of one channel at a time. Removal or bypass of one signal channel place that circuit in the half-tripped mode.

Hydrostatic tests of isolation valves in series are performed by first testing the upstream valve with the second valve open, then opening the upstream valve and closing the second valve, so that each valve will have an independent test.

The main steam and feedwater barriers and isolation valves in systems that connect to the reactor coolant system are hydrostatically tested to measure leakage.

Valves in the residual heat removal system are not considered to be isolating valves in the usual sense inasmuch as the system would be in operation under accident conditions.

Field and operational inspection and testing have been divided into three phases:

1. Construction tests; those taking place during erection of the containment building liner.

2. Preoperational tests; those taking place after the containment structure was erected and all penetrations were complete and installed.

3. Postoperational tests; monitoring during reactor operation.

5.1.10 Construction Tests

During erection of the liner, the following inspection and tests were performed.

5.1.10.1 Bottom Liner Plates

All liner plate welds are tested for leaktightness by vacuum box. The box is evacuated to at least a 5 psi pressure differential with the atmospheric pressure.

After completion of a successful leak test, the welds were covered by channels. A strength test was performed by applying 54 psig air pressure to the channels in the zone for a period of 15 min.

The zone of channel covered welds was pressurized to 47 psig with a 20-percent by weight of freon-air mixture. The entire run of the channel-to-plate welds was then traversed with a halogen leak detector.

The sensitivity of the leak detector was 1 x 10^{-9} standard cc per second. The sniffer was held approximately 0.5-in. from the weld and traversed at a rate of about 0.5-in./sec. The detection of any amount of halogen indicated a leak requiring weld repairs and retesting.

After the halogen test was completed, all liner welds not accessible for radiography were pressurized with air to 47 psig and soap-tested. Any leaks indicated by bubbles were repaired and retested. Where leaks occurred, welds were removed by arc gouging, grinding, chipping, and/or machining before rewelding. In addition, the zone of channels was held at the 47 psig air pressure for a period of at least 2 hr. The drop in pressure did not exceed the equivalent of a leakage of 0.05-percent of the containment building volume per day. Compensation for change in ambient air temperature was made.

5.1.10.2 Vertical Cylindrical Walls and Dome

For the liner, a complete radiograph was made of the first 10-ft of full penetration weld made by each welder or welding operation. A minimum of a 12-in. film "spot" radiograph was made every 50-ft of weld thereafter on the side walls and dome, except where backup plates were used. The radiograph films were reviewed by United Engineers and Constructors. When a spot radiograph showed defects that required repair, two adjacent spots were radiographed. If defects requiring repair were shown in either of these, all of the welding performed by the responsible operator or welder was 100-percent radiographed to determine the area of defect.

The performance and acceptance standards for all radiography were ASME Section VIII, Paragraph UW51.

The liner plate-to-plate welds were tested for leaktightness by vacuum box techniques. After successful completion of the spot radiography and vacuum box tests and subsequent repair of all defects, the channels were welded in place over all seam welds in a predetermined zone. A strength test was performed on the liner plate weld and the channel weld by pressurizing the channel with air at 54 psig for 15 min. In addition, each zone of channel covered weld was leak-tested using the freon-air mixture at 47 psig.

In locations where radiography was not possible, such as the lower courses of shell plates where backup plates were used, and where liner bottom welds and floor plate welds were made to angles and tees, the liner fabricator welded on a 2-in. long overrun coupon. The overrun coupon was chipped off, marked for location and given to United Engineers and Constructors for testing. These welds were also vacuum box tested.

Welded studs were visually inspected, and at least one at the beginning of each day's work and another at approximately mid-day were bend-tested to 45 degrees for each welder. Studs failing visual or bend-testing were removed.

While the liner is not a pressure vessel, industry experience has shown that leaks in pressure vessels normally occur at joints. For this reason, and following current liner fabrication practice, there was no radiographic or other nondestructive examination of liner plate.

5.1.10.3 Penetrations

Strength and leak tests of individual penetration internals and closures and sleeve weld channels were performed in a similar manner to the above and all leaks repaired and the penetration or weld channel retested until no further leaks were found. See Figures 5.1-53 through 5.1-56 for the areas of the containment and liner, which were instrumented for the strength test.

5.1.11 Preoperational Tests

All penetrations and the welds joining these penetrations to the containment liner and the liner seam welds were designed to provide a double barrier, which can be continuously pressurized at a pressure higher than the design pressure of the containment. This blocks all of these potential sources of leakage with a pressurized zone and at the same time provides a means of monitoring the leakage status of the containment, which is more sensitive to changes in the leakage characteristics of these potential leakage sources.

After the containment building was complete with liner, concrete structures, and all electrical and piping penetrations, equipment hatch and personnel locks were in place, the following tests were performed.

5.1.11.1 Strength Test

A pressure test was made on the completed building using air at 54 psig. This pressure was maintained on the building for a period of at least 1 hr. During this test, measurements and observations were made to verify the adequacy of the structural design. For a description of observations, cracks, strain gauges, etc., refer to Reference 18.

5.1.11.2 Integrated Leakage Rate Test: (Type A)

The integrated leakage rate tests were performed on the containment building at 47 psig using the absolute method. This leakage test was performed with the double penetration and weld channel zones open to the containment atmosphere. The leakage rate demonstrated by this test was equal to or less than 0.1-percent of the containment free volume per day at design basis accident conditions. After it was assured that there were no defects remaining from construction, a sensitive leak rate test was conducted.

5.1.11.3 Sensitive Leak Rate Test: (Type B)

The sensitive leak rate test included only the volume of the weld channels and double penetrations. This test was considered more sensitive than the integrated leakage rate test, as the instrumentation used permitted a direct measurement of leakage from the pressurized zones. The sensitive leak rate test was conducted with the penetrations and weld channels at 50 psig and with the containment building at atmospheric pressure. The leak rate for the double penetrations and weld channel zones was equal to or less than 0.2-percent of the containment free volume per day.

5.1.11.4 Containment Isolation Valve Test: (Type C)

These tests were conducted to detect leaks through certain containment isolation valves.

5.1.12 Postoperational Tests

Containment testing is conducted in accordance with the Technical Specifications and 10 CFR Appendix J, including integrated leakage rate tests at the containment design pressure, In 1997, the Technical Specifications were amended to allow the use of 10 CFR 50 Appendix J, Option B (as modified by approved exemptions) and NRC Regulatory Guide 1.163 dated September 1995 for integrated leakage rate tests, air lock tests, and containment isolation valve operability tests.

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TABLE 5.1-1
Flooded Weights - Containment Building

Item	Flooded Operating Weight, lbs
Pressurizer -1	346,000
Steam generators - 4	3,746,000
Reactor - 1	
(a) Vessel	868,000
(b) Internals	420,000
(c) Piping	1,000,000
Reactor pumps - 4	824,000
Accumulator tanks - 4	529,000
175-ton polar crane - 1	650,000
Ventilation fans - 5	656,000
Reactor coolant drain tank - 1	20,000
Pressure relief tank - 1	100,000
Other miscellaneous equipment	100,000
Total	9,259,000

TABLE 5.1-2
Containment Liner Insulation Properties

1. Elevation 46-ft to 64-ft liner insulation:

   - 1-1/4-in. polyvinylchloride insulation, Vinylcel, as manufactured by Johns-Mansville and 1-1/2 in. Pittsburgh Corning Foamglass Insulation.
   - 0.019-in. thick stainless steel jacket (exposed side) except for areas using Pittsburgh Corning Foamglass Insulation in which a jacket thickness of 0.024" is used.
   - Insulation adhesive is Johns-Manville Dutch Brand FN12 or an approved equal.

2. Elevation 64-ft to 89-ft liner insulation:[1]

   - 1.5-in. thick FOAMGLAS® with density of 8.5 to 9 lb/ft^3, as manufactured by Pittsburgh Corning Corporation. This insulation has a thermal conductivity of 0.5 - 0.525 $BTU\text{-}in/hr\text{-}ft^2\text{-}°F$ and a specific heat (Cp) of 0.18 BTU/lb-°F.
   - 1/16-in. commercial grade pure asbestos paper backing adjacent to the liner plate on the unexposed face.
   - The adhesive bonding the FOAMGLAS® to the asbestos paper is Cadoprene No. 434 and bonding the stainless steel jacket to the FOAMGLAS® is Cadoseal No. 700 by Epolux Manufacturing Corporation.

Note:
[1] Insulation from Elevation 82-ft to 89-ft only covers part of the circumferential area in the north and south-southwest quadrants.

5.1 FIGURES

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Figure 5.1-5	Containment Building General Arrangement Elevation - Sheet 1 Replaced with Plant Drawing 9321-2506
Figure 5.1-6	Containment Building General Arrangement Elevation - Sheet 2 Replaced with Plant Drawing 9321-2507
Figure 5.1-7	Containment Building General Arrangement Elevation - Sheet 3 Replaced with Plant Drawing 9321-2508
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Figure 5.1-9	Deleted

Figure 5.1-10	Deleted
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5.2 CONTAINMENT ISOLATION SYSTEM

5.2.1 Design Basis

Each system whose piping penetrates the containment leakage limiting boundary is designed to maintain or establish isolation of the containment from the outside environment under the following postulated conditions:

1. Any accident for which isolation is required (severely faulted conditions) with

2. A coincident independent single failure or malfunction (expected fault condition) occurring in any active system component within the isolated bounds.

Piping penetrating the containment is designed for pressures at least equal to the containment design pressure. Containment isolation valves are provided as necessary in lines penetrating the containment to ensure that no unrestricted release of radioactivity can occur. Such releases might be due to rupture of a line within the containment concurrent with a loss-of-coolant accident, or due to rupture of a line outside the containment that connects to a source of radioactive fluid within the containment.

In general, isolation of a line outside the containment protects against rupture of the line inside concurrent with a loss-of-coolant accident, or closes off a line, which communicates with the containment atmosphere in the event of a loss-of-coolant accident.

Isolation of a line inside the containment prevents flow from the reactor coolant system or any other large source of radioactive fluid in the event that a piping rupture outside the containment occurs. A piping rupture outside the containment at the same time as a loss-of-coolant accident is not considered credible, as the penetrating lines are seismic Class I design up to and including the second isolation barrier and are assumed to be an extension of containment.

The isolation valve arrangement provides two barriers between the reactor coolant system or containment atmosphere and the environment.

System design is such that failure of one valve to close will not prevent isolation, and no manual operation is required for immediate isolation. Automatic isolation is initiated by a containment ventilation isolation signal, a Phase A isolation signal ("T" signal), a Phase B isolation signal ("P" signal), or manually. See Section 5.2.4 or Chapter 7.0 for further details.

The containment isolation valves have been examined to ensure that they are capable of withstanding the maximum potential seismic loads.

To ensure their adequacy in this respect:

1. Valves are located in a manner to reduce the accelerations on the valves. Valves suspended on piping spans are reviewed for adequacy for the loads to which the span would be subjected. Valves are mounted in the position recommended by the manufacturer.

2. Valve yokes are reviewed for adequacy and strengthened as required for the response of the valve operator to seismic loads.

3. Where valves are required to operate during seismic loading, the operator forces are reviewed to ensure that system function is preserved. Seismic forces on the operating parts of the valve are small compared to the other forces present.

4. Control wires and piping to the valve operators are designed and installed to ensure that the flexure of the line does not endanger the control system. Appendages to the valve, such as position indicators and operators, are checked for structural adequacy.

Isolation valves are provided as necessary for all fluid system lines penetrating the containment to ensure at least two barriers for redundancy against leakage of radioactive fluids to the environment in the event of a loss-of-coolant accident. These barriers, in the form of isolation valves or closed systems, are defined on an individual line basis. In addition to satisfying containment isolation criteria, the valving is designed to facilitate normal operation and maintenance of the systems and to ensure reliable operation of other engineered safeguards systems.

With respect to numbers and locations of isolation valves, the criteria applied are generally those outlined by the seven classes described in Section 5.2.2 below.

5.2.2 System Design

The seven classes listed below are general categories into which lines penetrating containment may be classified. The seal water referred to in the listing of categories is provided by the isolation valve seal water system described in Section 6.5. The following notes apply to these classifications.

1. The "not-missile-protected" designation refers to lines that are not protected throughout their length inside containment against missiles generated as the result of a loss of coolant accident. These lines, therefore, are not assumed invulnerable to rupture as a result of a loss of coolant.

2. In order to qualify for containment isolation, valves inside the containment must be located behind the missile barrier for protection against loss of function following an accident.

3. Manual isolation valves that are locked closed or otherwise closed and under administrative control during power operation qualify as automatic trip valves.

4. A check valve qualifies as an automatic trip valve in certain incoming lines not requiring seal water injection.

5. The double disc type of gate valve is used to isolate certain lines. When sealed by water injection, this valve provides two barriers against leakage of radioactive liquids or containment atmosphere.

6. In lines isolated by globe valves and provided with seal water injection, the valves are generally installed so that the seal water wets the stem packing.

7. Loss of seal water through those isolation valves closed only by a containment isolation phase B signal is prevented by solenoid operated valves in the seal water injection lines. Excessive loss of seal water through motor operated isolation valves that could fail to close in response to a containment isolation signal is limited by flow restrictive orifices installed in the seal water lines. A water seal at the failed valve is ensured by proper slope of the protected line, or a loop seal, or by additional valves on the side of the isolation valves away from the containment.

8. Isolated lines between the containment and the second outside isolation valve are designed to the same seismic criteria as the containment vessel and are considered to be an extension of containment.

5.2.2.1 Class 1, Outgoing Lines, Reactor Coolant System

Outgoing lines connected to the reactor coolant system that are normally or intermittently open during reactor operation are provided with at least two automatic trip valves in series located outside the containment. Automatic seal water injection is provided for lines in this classification.

An exception to the general classification is the residual heat removal loop's reactor coolant system suction line, which has two barriers established by normally closed valves located outside containment.

5.2.2.2 Class 2, Outgoing Lines

Outgoing lines not connected to the reactor coolant system that are normally or intermittently open during operation and not missile protected or that can otherwise communicate with the containment atmosphere following an accident are provided at a minimum with two automatic trip valves in series or a single automatic double-disc gate valve outside containment. Automatic seal water injection is provided for lines in this classification. Most of these lines are not vital to plant operation following an accident.

5.2.2.3 Class 3, Incoming Lines

Incoming lines connected to open systems (i.e., systems that are in some way connected to the containment environment) outside containment, and not missile protected or that can otherwise communicate with the containment atmosphere following an accident are provided with one of the following arrangements outside containment:

1. Two automatic trip valves in series, with automatic seal water injection. This arrangement is provided for lines that are not necessary to plant operation after an accident.

2. Two manual isolation valves in series, with manual seal water injection. This arrangement is provided for lines that remain in service for a time, or are used periodically, subsequent to an accident.

Incoming lines connected to closed systems outside containment, and not missile protected or that can otherwise communicate with the containment atmosphere, are provided either with two isolation valves in series outside containment with seal water injection, or at a minimum, with one check valve or normally closed isolation valve located either inside or outside containment. The closed piping system outside containment provides the necessary isolation redundancy for lines that contain only one isolation valve.

Exception is the containment spray headers, for which valving is based on safeguards requirements.

5.2.2.4 Class 4, Missile Protected Lines

Incoming and outgoing lines that penetrate the containment and that are normally or intermittently open during reactor operation and are connected to closed systems inside the containment and protected from missiles throughout their length, are provided with at least one manual isolation valve located outside the containment. Seal water injection is not required for this class of penetration. An exception is the residual heat exchanger cooling water lines for which design is based on safeguards requirements.

5.2.2.5 Class 5, Normally Closed Lines Penetrating the Containment

Lines that penetrate the containment and that can be opened to the containment atmosphere but that are normally closed during reactor operation are provided with two isolation valves in series or one isolation valve and one blind flange.

5.2.2.6 Class 6, Special Service Lines

There are a number of special groups of penetrating lines and containment access openings. These are discussed below.

Each ventilation purge duct penetration is provided with two tight-closing butterfly valves, which are normally closed during reactor power operation and are actuated to the closed position automatically upon a containment isolation or a containment high radiation signal. One valve is located inside and one valve is located outside the containment at each penetration. The space between valves is pressurized by air from the weld channel and penetration pressurization system whenever they are closed. Blind flanges can also be used for containment isolation in place of automatic purge isolation valves, provided they meet the same design criteria as the isolation valves.

The containment pressure relief line is similarly protected. However, because the line is periodically opened during reactor power operation, three tight closing butterfly valves in series are provided, one inside and two outside the containment. These valves also are actuated to the closed position upon a containment isolation or containment high radiation signal. The two intravalve spaces are pressurized by air from the weld channel and penetration pressurization system whenever they are closed.

The equipment access closure is a bolted, gasketed closure that is air sealed during reactor operation. The personnel air locks consist of two doors in series with mechanical interlocks to ensure that one door is closed at all times. Each air lock door and the equipment closure are provided with double gaskets to permit pressurization between the gaskets by the weld channel and penetration pressurization system. (See Section 6.6.)

The fuel transfer tube penetration inside the containment is designed to present a missile-protected and pressurized double barrier between the containment atmosphere and the atmosphere outside the containment. The penetration closure is treated in a manner similar to the equipment access hatch. A positive pressure is maintained between the double gaskets of the tube cover flange to establish the double barrier between the containment atmosphere and the inside of the fuel transfer tube. The interior of the fuel transfer tube is not pressurized. Seal water injection is not required for this penetration.

The following lines would be subjected to pressure in excess of the isolation valve seal water system pressure (~52 psig) in the event of an accident, due to operation of the safety injection system recirculation pumps:

1. Residual heat removal loop inlet line.
2. Residual heat removal loop outlet line.
3. Bypass line from residual heat exchanger outlet to safety injection pumps suction.
4. Residual heat removal pumps mini-flow line.
5. Residual heat removal loop sample line.
6. Recirculation pump discharge sample line.

These lines are isolated by double disk gate valves or double valves, which can be sealed by nitrogen gas from the high pressure nitrogen supply of the isolation valve seal water system. A self-contained pressure regulator operates to maintain the nitrogen injection pressure slightly higher than the maximum expected line pressure. These valves are closed or intermittently operated during reactor operation, and the nitrogen gas injection is manually initiated.

Lines, which are capable of communicating with the containment atmosphere (normally filled with air or vapor) include:

1. Steam jet air ejector return line to containment.
2. Containment radiation monitor inlet and outlet lines.

In an accident condition the space between the two containment isolation valves in each line are sealed by pressurizing with air from the weld channel and penetration pressurization system. The air is introduced into each space at approximately 2 psi above the containment design pressure through a separate line from the weld channel and penetration pressurization system. Parallel, redundant, fail-open valves in each injection line open on the appropriate containment isolation signal to provide a reliable supply of pressurizing air. A flow-limiting orifice in each injection line prevents excessive air consumption if one of these valves spuriously fails to open, or if one of the containment isolation valves fails to respond to the "trip" signal.

5.2.2.7 Class 7, Steam and Feedwater Lines

These lines and the shell side of the steam generator are considered basically as an extension of the containment boundary and as such must not be damaged as a consequence of reactor coolant system damage. This requires that the steam generator shell, feed and steam lines within the containment are to be classified and designed for the reactor coolant system missile-protected category. The reverse is also true in that a steam break is not to cause damage to the reactor coolant system.

5.2.3 Isolation Valves And Instrumentation Diagrams

Figures 5.2-1 through 5.2-28 show all valves in lines leading to the atmosphere or to closed systems on both sides of the containment barrier, valve actuation and preferential failure modes, the application of "trip" (containment isolation) signals, relative location of the valves with respect to missile barriers, and the boundaries of seismic Class I designed lines. The item numbers in these figures align with the item numbers in Table 5.2-1. Figure 5.2-29 defines the nomenclature and symbols used.

5.2.4 Valve Parameters Tabulation

A summary of the fluid systems lines penetrating containment and the valves and closed systems employed for containment isolation is presented in Table 5.2-1. Each valve is described as to type, operator, position indication and open or closed status during normal operation, shutdown and accident conditions. Information is also presented on valve preferential failure mode, automatic trip by the containment isolation signal, and the fluid carried by the line.

Containment isolation valves are provided with actuation and control equipment appropriate to the valve type. For example, air-operated globe and diaphragm (Saunders Patent) valves are generally equipped with air diaphragm operators, with fail-safe operation provided by the control devices in the instrument air supply to the valve. Motor-operated gate valves are capable of being supplied from reliable onsite emergency power as well as their normal power source. Manual and check valves, of course, do not require actuation or control systems.

The containment isolation system is brought into service by one of three conditions: phase A isolation signal, phase B isolation signal, or containment ventilation isolation signal.

The automatically tripped isolation valves are actuated to the closed position by any of these isolation signals. The first of these signals is derived in conjunction with safety injection actuation, and trips the majority of the automatic isolation valves. These are valves in the so-called "nonessential" [***Note*** *- "Nonessential" process lines are defined as those, which do not increase the potential for damage to in-containment equipment when isolated. "Essential" process lines are those providing cooling water and seal water flow for the reactor coolant pumps. These services should not be interrupted unless absolutely necessary while the reactor coolant pumps are operating.*] process lines penetrating the containment. This is defined as "phase A" isolation and the trip valves are designated by the letter "T" in the isolation diagrams, Figures 5.2-1 through 5.2-29. This signal also initiates automatic seal water injection (See Section 6.5). The second, or "phase B", containment isolation signal is derived upon actuation of the containment spray system, and trips the automatic isolation valves in the so-called "essential"* process lines penetrating the containment. These trip valves are designated by the letter "P" in the isolation diagrams. Containment ventilation isolation represents closing of the three ventilation lines to the containment and will be automatically activated by high containment radioactivity, a phase A isolation signal, or automatic containment spray (and associated phase B) actuation; see Section 5.3 for further information on the containment, heating, cooling and ventilation system.

A manual containment isolation signal can be generated from the control room for either phase A or phase B isolation. These signals perform the same functions as the automatically derived signals. The containment ventilation isolation signal can be manually activated by a manual safety injection signal, a manual phase A containment isolation signal, or a manual containment spray signal.

Non-automatic isolation valves, i.e., remote stop valves and manual valves, are used in lines, which must remain in service, at least for a time, following an accident. These are closed manually if and when the lines are taken out of service.

Standard closing times available with commercial valve models are adequate for the sizes of containment isolation valves used. Valves equipped with air-diaphragm operators generally close in approximately 2 sec. The typical closing time available for large motor-operated gate valves is 10 sec. Closing times of greater than 10 seconds are permitted on a case by case basis if properly justified by an individual valve evaluation.

The large butterfly valves used to isolate the containment ventilation purge ducts are equipped with air piston operators and spring returns capable of closing the valves. The butterfly valves used to isolate the 10-in. pressure relief line are equipped with air piston operators each with a separate accumulator air supply on each valve capable of closing the valves. These valves all fail to the closed position on loss of control signal or instrument air. Allowable closure time for these valves is less than or equal to 3 seconds.

5.2.5 Valve Operability

All containment isolation valves, actuators, and controls are located so as to be protected against missiles that could be generated as the result of a loss-of-coolant accident. Only valves so protected are considered to qualify as containment isolation valves.

Only isolation valves located inside containment are subject to the high-pressure, high-temperature, steam-laden atmosphere resulting from an accident. Operability of these valves in the accident environment is ensured by proper design, construction, and installation, as reflected by the following considerations:

1. All components in the valve installation, including valve bodies, trim and moving parts, actuators, instrument air and control and power wiring, are constructed of materials sufficiently temperature resistant to be unaffected by the accident environment. Special attention is given to electrical insulation, air operator diaphragms and stem packing material.

2. In addition to normal pressures, the valves are designed to with-stand maximum pressure differentials in the reverse direction imposed by the accident conditions.

This criterion is particularly applicable to the butterfly-type isolation valves used in the containment purge lines. Valve actuators are installed on these butterfly valves and travel is limited to a maximum of 60 degrees to ensure that the valves will be able to close against the maximum calculated design-basis accident pressure of 47 psig. An adjustable position setting on the actuators allows the valves to be opened to a full 90-degree position when containment integrity is not required.

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 1 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method[1]	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
1	Pressurizer relief tank to gas analyzer RCS	5.2-1	549	Globe	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)	No	G	Hot	
			548	Globe	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)				
2.	Pressurizer relief tank N_2 supply tank RCS	5.2-1	518	Check	-	No	Closed	Closed	Closed	-	-	-	Yes/No	G	Cold	
			3418	Globe	Sole.	No	Open	Open	Op/Cl	FC	-	-				
			3419	Globe	Sole.	No	Open	Open	Op/Cl	FC	-	-				
			4136	Dia.	Manual	No	Closed/OI	Closed/OI	Op/Cl	-	-	-				
3.	Pressurizer relief tank makeup - RCS	5.2-1	552	Dia	Air	Yes	Closed	Closed	Closed	FC	T	Water (A)	No	W	Cold	
			519	Dia	Air	Yes	Closed	Closed	Closed	FC	T	Water (A)				
4.	Residual heat removal return - ACS/SIS	5.2-2	$741A_{1A}$	Check	-	No	Closed	Open	Op/CL	-	-	RHR	Yes	W	Hot	
			744_{1A}	DDV	Motor	Yes	Open	Open	Op/CL	FAI	-	Nit (M)				May be closed depending on accident condition
5.	Resid. Heat removal loop to - S.I.pumps - ACS/SIS	5.2-2	888A	DDV	Motor	Yes	Closed/OI	Closed	Op/Cl	FAI	-	Nit (M)	Yes	W	Hot**	
			888B	DDV	Motor	Yes	Closed/OI	Closed	Op/Cl	FAI	-	Nit (M)				
	To sampling system ACS/SS	5.2-2	958	Globe	Motor	No	Closed/OI	Closed	Closed	FAI	-	Nit (M)	Yes	W	Hot	May be used during shutdown and after accident
			959	Globe	Motor	No	Closed/OI	Closed	Closed	FAI	-	Nit (M)	Yes	W		
			990D	Globe	Manual	No	LC/OI	LC	LC	-	-	Nit (M)	No	-	-	
	RHR pump mini-flow line	5.2-2	1870	Globe	Motor	Yes	Open	Open	Op/Cl	FAI	-	Nit (M)	Yes/No	W	Hot	
			743	Globe	Motor	Yes	Open	Open	Op/Cl	FAI	-	Nit (M)	Yes/No	W		

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 2 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method1	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
6.	Residual heat removal loop-out - ACS	5.2-2	732 1A	DDV	Manual	No	LC/OI	Open	Closed	-	-	Nit (M)	No	W	Hot	
7.	Containment sump recirculation - ACS/SIS	5.2-2	885A 885B	DDV 2A DDV 2A	Motor Motor	Yes Yes	Closed/OI Closed/OI	Closed Closed	Closed 2 Closed 2	FAI FAI	- -	RHR RHR	No 2	W	Cold	2. Normally closed but may be opened after accident if normal recirculation path from recirculation pump not available 2A. The upstream disc (nearest containment) of 885A and the downstream disc (RHR Loop side) of 885B have a 3/16" hole to prevent pressure locking
8.	Letdown line - CVCS	5.2-3	201 202	Globe Globe	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Hot	
9.	Charging line - CVCS	5.2-3	205 226 227	Gate Globe Globe	Motor Motor Motor	No No No	Open Open Closed/OI	Open Open Closed	Op/Cl Op/Cl Op/Cl	FAI FAI FAI	- - -	Water (M) Water (M) Water (M)	Yes* Yes* Yes*	W	Cold	* May be used depending on accident.
10.	Reactor coolant pump seal-water supply lines (4) - CVCS	5.2-4	250ABCD 4925, 4926, 4927, 4928	Globe Globe	Motor Motor	No No	Open Open	Open Open	Op/Cl Op/Cl	FAI FAI	- -	Water (M) Water (M)	Yes 3 Yes 3	W W	Cold Cold	3. Manual isolation if and when pumps are stopped.
11.	Reactor coolant pump seal water return - CVCS	5.2-4	222	DDV	Motor	Yes	Open	Open	Closed	FAI	P	Water (A)	No	W	Cold	
12.	Reactor coolant sample line - SS	5.2-5	956E 956F	Globe Globe	Motor Motor	Yes Yes	Op/Cl Op/Cl	Op/Cl Op/Cl	Closed Closed	FAI FAI	T T	Water (A) Water (A)	Yes 4	W	Hot	4. Used to take postaccident RCS samples
13.	Fuel transfer tube - FHS	5.2-5	A	Blind flange	-	No	Closed	-	-	-	-	Air 5	No	W	Cold	Flange is double gasketed in refuel-ing canal (missile protected). 5. Normally seal with air (WCPPS)

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 3 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Posit. Isolation Trip	Testing/ Sealing Method[1]	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
14.	Containment spray headers (2) - SIS	5.2-6	869A,B 867A,B 878A	DDV Check Globe	Motor - Manual	No No No	Open Closed LC/OI	Op/Cl Closed Closed	Op/Cl Op/Cl Closed	FAI - -	- - -	Water (M) - -	Yes	W	Cold	
15.	Safety injection headers (2) - SIS	5.2-7	850A 851A 851B 850B	DDV DDV DDV DDV	Motor Motor Motor Motor	No Yes Yes No	Open Open Open Open	Open Open Open Open	Op/Cl Op/Cl Op/Cl Op/Cl	FAI FAI FAI FAI	- - - -	Water(M) Water(M) Water(M) Water(M)	Yes Yes Yes Yes	W W W W	Hot** Hot** Hot** Hot**	
16.	Safety injection test line - SIS	5.2-7	859A 859C	Globe Globe	Manual Manual	No. No	LC/OI LC/OI	Closed Closed	Closed Closed	- -	- -	Water (A) Water (A)	No	W	Cold	
17.	Accumulator/ OPS N_2 supply - SIS	5.2-8	4312 863	Check Globe	- Air	No Yes	Closed Op/Cl	Closed Op/Cl	Closed[6] Closed[6]	- FC	- T	- -	No No	G	Cold	6. Could be opened depending on type of accident
18.	Accumulator sample - SS	5.2-8	956G 956H	Globe Globe	Air Air	Yes Yes	Op/Cl Op/Cl	Op/Cl Op/Cl	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Cold	Valves A and B opened intermittently to take sample
19.	Primary system vent header and N_2 supply line - WDS	5.2-9	1786 1787 3416 3417 5459 1616	Dia Dia Globe Globe Dia Check	Air Air Sole. Sole. Manual -	Yes Yes No No No No	Open Open Open Open Closed/OI Closed	Closed Closed Open Open Closed Closed	Closed Closed Op/Cl Op/Cl Op/Cl Op/Cl	FC FC FC FC - -	T T - - - -	Water (A) Water (A) - - - -	No Yes/No	G G	Hot Hot	
20.	Reactor coolant drain tank to gas analyzer - WDS	5.2-9	1788 1789	Dia Dia	Air Air	Yes Yes	Op/Cl Op/Cl	Op/Cl Op/Cl	Closed Closed	FC FC	T T	Water (A) Water (A)	No	G	Hot	Valves opened intermittently
21.	RCDT pump discharge – WDS	5.2-9	1702 1705	Dia Dia	Air Air	Yes Yes	Open Open	Op/Cl Op/Cl	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Cold	Valves open intermittently

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 4 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method1	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
22.	Reactor coolant pump cooling water in - ACS	5.2-10	797	DDV	Motor	Yes	Open	Op/Cl	Closed	FAI	P	Water (A)	No7	W	Cold	7. Could be used depending on the type of accident
23.	Reactor coolant pump water out (6”) - ACS	5.2-10	784	DDV	Motor	Yes	Open	Op/Cl	Closed	FAI	P	Water (A)	No8	W	Cold	8. Could be used depending on the type of accident
24.	Reactor coolant pump water out (3”) - ACS	5.2-10	FCV-625	DDV	Motor	Yes	Open	Op/Cl	Closed	FAI	P	Water (A)	No9	W	Cold	9. Could be used depending on the type of accident
25.	Resid. Heat exch. Cooling water in - ACS	5.2-11	CS	-	-	-	-	-	-	-	-	-	Yes	W	Hot	Residual heat exchanger and associated component cooling lines are a missile protected closed system
			CS	-	-	-	-	-	-	-	-	-	Yes	W	Hot	Component cooling system closed
26.	Resid. Heat exch. Cooling water return - ACS	5.2-11	822A1B	Gate	Motor	Yes	Closed	Open	Open	FAI	-	-	Yes	W	Cold	
			822B1B	Gate	Motor	Yes	Closed	Open	Open	FAI	-	-	Yes	W	Cold	
			CS	-	-	-	-	-	-	-	-	-	Yes	W	Cold	Component cooling system closed
27.	Recir. Pump cooling water supply - ACS	5.2-12	753H1B	Gate	Manual	No	Open	Open	Op/Cl	-	-	-	Yes	W	Cold	May be closed depending on accident condition
			CS	-	-	-	-	-	-	-	-	-	Yes	W	Cold	Component cooling system closed
28.	Recir. Pump cooling heater return - ACS	5.2-12	753G1B	Gate	Manual	No	Open	Open	Op/Cl	-	-	-	Yes	W	Cold	May be closed depending on accident condition
			CS	-	-	-	-	-	-	-	-	-	Yes	W	Cold	Component cooling system closed
29.	Excess letdown heat exchanger cooling water in – ACS	5.2-13	791	Dia	Air	Yes	Closed	Closed	Closed	FC	T	Water (A)	No	W	Cold	
			798	Dia	Air	Yes	Closed	Closed	Closed	FC	T	Water (A)				

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 5 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method[1]	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
30.	Excess letdown heat exchanger cooling water out - ACS	5.2-13	796 793	Globe Dia	Air Air	Yes Yes	Closed Closed	Closed Closed	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Cold	
31.	Containment sump pump discharge – WDS	5.2-13	1728 1723	Dia Dia	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Cold	
31a.	Sampling system return - WDS	5.2-13	5132 4399	Globe Globe	Motor Motor	Yes Yes	Closed Closed	Closed Closed	Cl/Op Cl/Op	FAI FAI	T T	Water (A) Water (A)	No/Yes[10] No/Yes[10]	W W	Cold Cold	10. Can be used to return highly radioactive water to containment after post-accident analysis of sampling system.
32.	Containment air sample in - rad. mon.	5.2-14	PCV-1234 PCV-1235	Dia Dia	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Air (A) Air (A)	No[11] No[11]	G G	Cold Cold	11. May be opened for air sampling following accident when the containment pressure is below 5 psig
33.	Containment air sample out - rad. mon.	5.2-14	PCV-1236 PCV-1237	Dia. Dia.	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Air (A) Air (A)	No[12] No[12]	G G	Cold Cold	12. May be opened for air sampling following accident when the containment pressure is below 5 psig
34.	Air ejector discharge to containment sec sys	5.2-14	PCV-1229 PCV-1230	Globe Globe	Air Air	Yes Yes	Closed Closed	Closed Closed	Closed Closed	FC FC	T T	Air (A) Air (A)	No No	G G	Cold Cold	
35.	Main steam headers[13]	-	CS	-	-	-	-	-	-	-	-	-	-	-	Hot	Steam generators 13. (four penetrations)

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 6 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method[1]	Used After Acid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
36.	Main feedwater headers	-	CS	-	-	-	-	-	-	-	-	-	-	-	Hot	Steam generators (four penetrations)
37.	Steam generator blowdown/sample sec. sys.	5.2-15	PCV1214 PCV1215 PCV1216 PCV1217	Globe	Air	Yes	Open	Op/Cl	Closed	FC	T	Water (A)	No	W	Hot	*(four penetrations)
			PCV1214A PCV1215A PCV1216A PCV1217A	Globe	Air	Yes	Open	Op/Cl	Closed	FC	T	Water (A)	No	W	Hot	
38.	S.G. blowdown sample															System deleted
39.	Ventilation system water cooling water in – SWS_{14}	5.2-16	SWN-41 SWN-42 SWN-43	BV Relief Gate(2) Globe(3)	Motor - Manual	No No No	Open Closed LC/OI	Open Closed Closed	Op/Cl Closed Op/Cl	FAI - -	- - -	SWS SWS SWS	Yes - Yes	W - -	Cold - -	Fan cooler units - missile protected, closed system 14. (five penetrations)
40.	Ventilation system motor cooling water out -	5.2-16	SWN-44	BV	Motor	No	Open	Open	Op/Cl	FAI	-	SWS	Yes	W	Cold	Fan cooler units - missile protected, closed system
	SWS_{15}	5–2–16	SWN-51	Globe	Motor	No	Open	Open	Op/Cl	FAI	-	SWS	Yes	W	Cold	15. (Five penetrations)
40a.	Ventilation system motor cooling water out - SWS_{16}	5–2–16	SWN-71	Globe	Motor	No	Open	Open	Op/Cl	FAI	-	SWS	Yes	W	Cold	16. Five penetrations
41.	Service air	5.2-17	SA-24 SA-24-1	Dia Dia	Manual Manual	No No	LC/OI LC/OI	LC LC	LC LC	- -	- -	Water (A) Water (A)	No	G	Cold	
42.	Not assigned															

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 7 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing $Method_1$	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
43.	Weld channel pressurization air supply $(PPS)_{17}$	5.2-17	$PCV1111\text{-}1_{1B}$ $PCV1111\text{-}2_{1B}$ CS	Ball Ball -	Manual - -	No - -	Open	Open	Op/Cl	- -	- -	- -	Yes -	G -	Cold -	17. Two penetrations penetration press system
44.	Spare	5.2-17	580A 580B	Needle Needle	Manual Manual	No No	LC/OI LC/OI	LC LC	LC LC	- -	- -	- -	No	G	Cold	Penetration capped inside containment and outside containment downstream of valve 580B
45.	Auxiliary steam supply	5.2-18	UH-43	DDV	Manual	No	LC/OI	$Closed_{18}$	LC	-	-	Water (A)	No	G	Hot	18. May be opened during shutdown for cont. heating
46.	Auxiliary steam supply condensate return	5.2-18	UH-44	DDV	Manual	No	LC/OI	$Closed_{19}$	LC	-	-	Water (A)	No	W	Hot	19. May be opened during shutdown for cont. heating
47.	City water	5.2-18	MW-17 MW-17-1	Gate Gate	Manual Manual	No No	LC/OI LC/OI	$Closed_{20}$ $Closed_{20}$	LC LC	- -	- -	Water (A) Water (A)	No	W	Cold	20. May be opened during shutdown for maintenance or fire protection purposes
48.	Purge supply duct in - vent. sys.	5.2-19	FCV-1170 FCV-1171	BV BV	Air Air	Yes Yes	$Closed_{21}$ $Closed_{21}$	Open Open	Closed Closed	FC FC	CVI CVI	Air (A) Air (A)	No	G	Cold	21. May be open for safety related purging, or to facilitate safety related surveillance or maintenance.
49.	Purge exhaust duct out - vent. sys.	5.2-19	FCV-1172 FCV-1173	BV BV	Air Air	Yes Yes	$Closed_{22}$ $Closed_{22}$	Open Open	Closed Closed	FC FC	CVI CVI	Air (A) Air (A)	No	G	Cold	22. May be open for safety related purging, or to facilitate safety related surveillance or maintenance.
50.	Containment pressure relief - vent	5.2-19	PCV-1190 PCV-1191 PCV-1192	BV BV BV	Air Air Air	Yes Yes Yes	$Closed_{23}$ $Closed_{23}$ $Closed_{23}$	Closed Closed Closed	Closed Closed Closed	FC FC FC	CVI CVI CVI	Air (A) Air (A) Air (A)	No	G	Cold	23. Opened intermittently for pressure relief.
51.	Recirculation pump discharge sample line	5.2-20	990A 990B	Globe Globe	Motor Motor	Yes Yes	Closed Closed	Closed Closed	Op/Cl Op/Cl	FAI FAI	T T	Nit (M) Nit (M)	No/Yes_{24}	W	Cold	24. Used periodically after accident to sample recirculation fluid.

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 8 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method[1]	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
52.	Pressurizer steam space sample line	5.2-20	956A	Globe	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)	No/Yes[25]	W	Hot	25. Could be used for taking postaccident samples.
			956B	Globe	Air	Yes	OP/Cl	OP/Cl	Closed	FC	T	Water (A)	No/Yes[25]	W	Hot	
53.	Pressurizer liquid space sample line	5.2-20	956C	Globe	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)	No/Yes[26]	W	Hot	26. Could be used for taking postaccident samples.
			956D	Globe	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)	No/Yes[26]	W	Hot	
54.	Containment	5.2-21	1814A	Globe	Manual	No	LO	Open	Op/Cl	-	-	-	Yes	G	Cold	
55.	Pressure		1814B													
56.	Instrumentation		1814C													
			CS	-	-	-	-	-	-	-	-	-	-	-	-	
57.	Postaccident containment sampling system supply and return lines (7)	5.2-22	SOV-5018	Globe	Sole.	Yes	Closed	Closed	Both[26a]	FC	-	-	Yes	G	Cold	26a. Isolation valves are opened intermittently after an accident.
			SOV-5020	Globe	Sole.											
			SOV-5022	Globe	Sole.											
			SOV-5024	Globe	Sole.											
			SOV-5019	Globe	Sole.	Yes	Closed	Closed	Both[26a]	FC	-	-	Yes	G	Cold	
			SOV-5021	Globe	Sole.											
			SOV-5023	Globe	Sole.											
			SOV-5025	Globe	Sole.											
58.	Spare	5.2-23														
59.	Spare	5.2-24														
60.	Spare															
61.	Spare															
62.	Spare															
63.	Not assigned															

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 9 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method[1]	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
64.	Instrument/air postaccident venting supply	5.2-25	IA-39	Check	-	No	Open	Open	Both	-	-	-	Yes/No[27]	G	Cold	27. Could be used to resupply instrument air to containment post-accident.
			PCV1228	Dia	Air	Yes	Open	Open	Both	FC	T	-	Yes/No[27]	G	Cold	
65.	Postaccident venting exhaust line	5.2-25	E-2	Dia	Air	No	Closed/OI	Closed	Both	FC	-	Air (A)	Yes/No[28]	G	Cold	28. Could be used after accident if containment venting were deemed necessary.
			E-1	Dia	Air	No	Closed/OI	Closed	Both	FC	-	Air (A)	Yes/No[28]	G	Cold	
			E-3	Dia	Air	No	Closed/OI	Closed	Both	FC	-	Air (A)	Yes/No[28]	G	Cold	
			E-5	Dia	Air	No	Closed/OI	Closed	Both	FC	-	Air (A)	Yes/No[28]	G	Cold	
66.	Deleted															
67.	Containment leak test air line[30]	5.2-26	A	Blind Flange	-	No	Closed	Closed	Closed	-	-	-	No.	Gas	Cold	30. Two penetrations
			B	Blind Flange W/Test Conn.	-	No	Closed	Closed	Closed	-	-	-	No	Gas	Cold	
68.	Equipment access	-	CS	-	-	-	-	-	-	-	-	Air (A)	No	-	-	
69.	Personnel air lock (2)	5.2-27	85A, 95A[31]	Ball	Interlk	Yes	Closed	Closed	Closed	-	-	Air (A)	No	Gas	Cold	31. 85A & 95A may be open when 85B & 95B are closed. 85B & 95B may be open when 85A & 95A are closed.
			85B, 95B[31]	Ball	w/door							Air (A)				
			85C, 95C	Spring check	-	No	Closed	Closed	Closed	-	-	-	No	Gas	Cold	
			85D, 95D	Spring check	-	No	Closed	Closed	Closed	-	-	-	No	Gas	Cold	
70.	Steam generator level, pressurizer level, and pressure pneumatic indication lines (4)	5.2-28	IIP-500	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes[32]	-	Cold	32. Depending on accident type
			IIP-501	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes[32]	-	Cold	
			IIP-502	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes[32]	-	Cold	
			IIP-503	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes[32]	-	Cold	
			IIP-504	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes[32]	-	Cold	
			IIP-505	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes[32]	-	Cold	
			IIP-506	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes[32]	-	Cold	
			IIP-507	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes[32]	-	Cold	

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 10 of 10)

Key:

FAI	- fail as is	RCS	- reactor coolant system	Hot – insulated & cooled penetration
FC	- fail closed	ACS	- auxiliary coolant system	Cold – standard piping or equipment penetration
FO	- fail open	WDS	- waste disposal system	Hot** - insulated non cooled penetration
LC	- locked closed	SIS	- safety injection system	IVSWS -isolation valve seal water system
LO	- locked open	SS	- sampling system	WCPPS -weld channel pressurization system
BV	- butterfly valve	CVCS	- chemical and volume control system	RHR -residual heat removal system
DDV	- double disk gate valve	Vent	- ventilation system	
Dia	- diaphragm valve	SWS	- service water system	
T	- containment isolation signal - phase A	FH	- fuel handling	
P	- containment isolation signal - phase B	PPS	- penetration pressurization system	
A	- automatic	CVI	- containment ventilation isolation signal	
M	- manual	CS	- closed system	
Op/Cl	- open/closed	Nit	- nitrogen	
OI	- may be opened intermittently to support plant operations			

Notes:

1. Sealing Methods and Test Pressures:
   - For valves sealed by IVSWS water (designated “Water (A)” or “Water (M)”), minimum test pressure is 52 psig.
   - For valves sealed by IVSWS nitrogen (designated “Nit (M)”), minimum test pressure is 47 psig.
   - For valves sealed by WCPPS (designated “Air (A)”), minimum test pressure is 47 psig.
   - For valves sealed by RHR system fluid (designated “RHR”), minimum test pressure is 52 psig (valves 741A,885A,885B).
   - For valves sealed by service water system (designated “SWS”), minimum test pressure is 52 psig (valve series SWN-41, SWN-42, SWN-43, SWN-44, SWN-51, SWN-71). Either the “A” or “B” valve(s) may serve as the required containment isolation valve(s) for the SWN-41, SWN-44 and SWN-71 series. Designation of the “B” valve(s) in the SWN-44 series requires the codesignation of the SWN-51 valves associated with the penetration(s) as an additional required containment isolation valve(s) (see Figure 5.2-16).
   - For all other isolation valves not sealed by a system, gas (ie. Nitrogen or air) is the test medium at a minimum pressure of 47 psig.

1A. These valves testable only when at cold shutdown (741A, 744, 732). In addition, according to the IPEC Containment Leakage Rate Testing Program (CLRTP) established in accordance with Technical Specification ITS 5.5.14, the line containing MOV-744 and its series check valve 741A does not represent a primary containment atmospheric leak pathway to the environs, given a single active failure. Therefore, they do not require Appendix J Type C local leak rate testing.

1B. These valves are excluded from Type C testing per License Amendment N0. 63, dated August 28, 1980 (822A, 822B, 753G, 753H, PCV-1111-1, PCV-1111-2).

1C. Containment Pressure Instrumentation (1814A, 1814B, 1814C) – LLRT is not required. Valve / penetration is open during Type A ILR Test.

5.2 FIGURES

Figure No.	**Title**
Figure 5.2-1	Containment Isolation System Penetration Schematics
Figure 5.2-2	Containment Isolation System Penetration Schematics
Figure 5.2-3	Containment Isolation System Penetration Schematics
Figure 5.2-4	Containment Isolation System Penetration Schematics
Figure 5.2-5	Containment Isolation System Penetration Schematics
Figure 5.2-6	Containment Isolation System Penetration Schematics
Figure 5.2-7	Containment Isolation System Penetration Schematics
Figure 5.2-8	Containment Isolation System Penetration Schematics
Figure 5.2-9	Containment Isolation System Penetration Schematics
Figure 5.2-10	Containment Isolation System Penetration Schematics
Figure 5.2-11	Containment Isolation System Penetration Schematics
Figure 5.2-12	Containment Isolation System Penetration Schematics
Figure 5.2-13	Containment Isolation System Penetration Schematics
Figure 5.2-14	Containment Isolation System Penetration Schematics
Figure 5.2-15	Containment Isolation System Penetration Schematics
Figure 5.2-16	Containment Isolation System Penetration Schematics
Figure 5.2-17	Containment Isolation System Penetration Schematics
Figure 5.2-18	Containment Isolation System Penetration Schematics
Figure 5.2-19	Containment Isolation System Penetration Schematics
Figure 5.2-20	Containment Isolation System Penetration Schematics
Figure 5.2-21	Containment Isolation System Penetration Schematics [Replaced with Plant Drawing 235296]
Figure 5.2-22	Containment Isolation System Penetration Schematics
Figure 5.2-23	Containment Isolation System Penetration Schematics
Figure 5.2-24	Containment Isolation System Penetration Schematics
Figure 5.2-25	Containment Isolation System Penetration Schematics
Figure 5.2-26	Containment Isolation System Penetration Schematics
Figure 5.2-27	Containment Isolation System Penetration Schematics
Figure 5.2-28	Containment Isolation System Penetration Schematics
Figure 5.2-29	Containment Isolation System Penetration Schematics

5.3 CONTAINMENT HEATING, COOLING AND VENTILATION SYSTEM

5.3.1 Design Basis

5.3.1.1 Performance Objectives

The containment heating, cooling and ventilation systems are designed to accomplish the following:

1. Remove the normal heat loss from all equipment and piping in the reactor containment during plant operation and to maintain a normal ambient temperature of 130°F or less.

2. Provide sufficient air circulation and filtering of iodine throughout all containment areas to permit safe and continuous access to the reactor containment within two hours after reactor shutdown assuming defects exist in 1-percent of the fuel rods.

3. Provide for positive circulation of air across the refueling water surface to assure personnel access and safety during shutdown.

4. Provide containment heating, if required, to assure a minimum containment ambient temperature of 50°F before the reactor is taken above the cold shutdown condition.

5. Provide for purging of the containment vessel to the plant vent for dispersion to the environment. The rate of release does not permit offsite dose to exceed Offsite Dose Calculation Manual (ODCM).

6. Provide for depressurization of the containment vessel following an accident. The postaccident design and operating criteria are detailed in Section 6.4.

7. Provide for continuous pressure relief via an exhaust system.

In order to accomplish these objectives the following systems are provided:

1. Containment recirculation cooling system
2. Control rod drive mechanism cooling system
3. Containment purge and pressure relief system
4. Containment auxiliary charcoal filter system
5. Steam heating system

5.3.1.2 Design Characteristics-Sizing

The design characteristics of the equipment required in the containment for cooling, filtration and heating to handle the normal thermal and air cleaning loads during normal plant operation are presented in Table 5.3-1. In certain cases where engineered safeguards functions also are served by the equipment, component sizing is determined from the heavier duty specifications associated with the design basis accident detailed further in Section 6.4.

5.3.2 System Design

5.3.2.1 Piping and Instrumentation Diagram

The containment ventilation, purging, and recirculation cooling and filtration systems flow diagram is shown in Plant Drawing 9321-4022 [Formerly UFSAR Figure 5.3-1]. The containment ventilation systems and main plant vent are designed as Class I structures.

5.3.2.2 Containment Cooling and Ventilation System

Air recirculation cooling during normal operation is accomplished using air handling units discharged into a common header ductwork distribution system to ensure adequate flow of cooled air throughout the containment. The cooling coils in each air handling unit transfer up to 61.7×10^6 Btu/hr in the event of an accident when supplied with approximately 1600 gpm cooling water at 95°F inlet temperature and steam-air flowrate of 64,500 cfm.

Each air-handling unit consists of the following equipment arranged so that, during normal and accident operation, air flows through the unit in the following sequence: cooling coils, moisture separators (demisters), centrifugal fan with direct-drive motor, and distribution header. The fans and motors of these units are equipped with vibration sensors to detect abnormal operating conditions in the early stages of the disturbance. The normal air flow rate per air-handling unit is approximately 70,000 cfm. Section 6.4.2 provides additional information on the operation of this system.

The following additional systems supplement the main containment recirculation system:

1. Control rod drive cooling system consisting of fans and ductwork to circulate air through the control rod drive mechanism shroud and discharge it to the main containment volume. Four direct driven axial flow fans are provided for use. There are two power supplies for each fan.

2. Two unit steam heaters are located in containment to provide additional area heating as required. The containment purge supply is also provided with steam pre-heating to supplement containment heating as required.

5.3.2.3 Containment Purge System

The containment purge system is independent of the primary auxiliary building exhaust system, (except for the common exhaust fans) and includes provisions for both supply and exhaust air. The supply system includes roughing filters, heating coils, fan, supply penetration with two butterfly valves for bubble tight shutoff, and a purge supply distribution header inside containment. The exhaust system includes exhaust penetration with two butterfly valves identical to those above, exhaust ductwork, filter bank with roughing, HEPA and charcoal filters, fans and exhaust vent. The purge supply and exhaust flow rates are nominally 23,000 cfm and 25,000 cfm respectively. The quick closing purge isolation valves close upon receipt of an accident signal. Allowable closure times for these valves are specified in Section 5.2.4.

During power operation, the purge system is routinely not operated. Prior to purging the containment air, particulate and gas monitor indications of the closed containment activity levels will be used to guide routine releases from the containment. During power operation, the containment air particulate and gas monitor indications will help determine desirability of using either one or both of two auxiliary particulate and charcoal filter units installed in the containment primarily for preaccess cleanup.

When containment purging is in progress for access following reactor shutdown, releases from the plant vent are continuously monitored with a gas monitor, as described in Section 11.2.

5.3.2.4 Purge System Isolation Valves

The purge supply and exhaust duct butterfly valves, both inside and outside the containment, are normally closed during power operation. They may be opened for safety related reasons, i.e., pressure control or to facilitate safety related surveillance or maintenance. The opening angle is limited during operation so that the valves can close against a differential pressure (see Section 5.2.5). The spaces between the closed valves are pressurized with air by the weld channel and penetration pressurization system. The valves are designed for rapid automatic closing by the containment isolation signal (derived from any safety injection signal), upon a signal of high activity level within the containment in the event of a radioactivity release when the purge line is open, or upon a manually initiated signal. To ensure optimum sealing of the resilient valve seats, the two valves located outside containment are enclosed and a minimum ambient temperature is maintained.

5.3.2.5 Containment Pressure Relief Line

The normal pressure changes in the containment during reactor power operation, and during plant cooldown if the containment purge system is not operating, will be handled by the containment pressure relief line. This line is equipped with three quick-closing butterfly type isolation valves, one inside and two outside the containment. The valves will be automatically actuated to the closed position by the containment isolation signal, by a containment high radioactivity signal, or upon a manually initiated signal. Each of these air operated valves is equipped with an accumulator to assure each can close even if the air supply is lost. The two intra-valve spaces are pressurized with air by the Weld Channel and Penetration Pressurization System when the valves are closed. The pressure relief line discharges to the plant vent. The opening angle of the pressure relief valves is limited during operation so the valves can close against a differential pressure.

5.3.2.6 Containment Purge and Pressure Relief Isolation Reset

Opening of the purge and pressure relief isolation valves following an isolation signal requires deliberate operator action by resetting all isolation signals and depressing both Containment Ventilation Isolation reset push buttons. Further, in order to reset the Containment Ventilation Isolation signal for Train B, the control switches for the purge and pressure relief isolation valves in Train B must first be placed in the closed position. In addition, guard plates are placed over the reset buttons. These three features preclude the possibility of inadvertently opening these valves.

REFERENCES FOR SECTION 5.3

1. Deleted

TABLE 5.3-1 (Sheet 1 of 2)
Containment Cooling and Ventilation System - Principal Component Data Summary

System	Units Installed	Unit Capacity	Units Required for Normal Operation
Containment cooling and recirculation			
Demister	5	72,500 cfm	5_2
Cooling coils - normal	5	2.20×10^6 Btu/hr$_1$	5_2
Cooling coils - DBA	5	61.7×10^6 Btu/hr$_3$	
Fans	5	72,500 cfm	5_2
Fan pressure	-	7.21-in. H_2O (Note 7)	
Fan motors (440 V, 3-phase)	5	350 hp	5_2
Control rod drive mechanism cooling			
Fans, standard conditions	4	15,000 cfm	3
Fan pressure	-	5.5-in. H_2O	
Fan motors	4	30 hp	3
Reactor compartment cooling			
Part of containment recirculation system	-	12,000 cfm	
Refueling canal air sweep			
Part of containment recirculation system	-	17,500 cfm	
Purge supply	1	23,000 cfm$_6$	Optional
Fan pressure	-	2.5-in. H_20	
Fan motors	1	40 hp	
25 psig steam preheat coils	1 set	3×10^6 Btu/hr	Optional
Air filters, roughing	-	-	1

TABLE 5.3-1 (Sheet 2 of 2)
Containment Cooling and Ventilation System - Principal Component Data Summary

System	Units Installed	Unit Capacity	Units Required for Normal Operation
Purge exhaust			
Fans, standard conditions	2[4]	55,500 cfm[6]	Optional
Fan pressure	-	10.3-in. H_2O	Optional
Fan motors	2	125 hp	Optional
Plenums	1	-[5]	Optional
HEPA filters	-	-	Optional
HECA filters (charcoal adsorbers)	1	-	Optional
Containment auxiliary charcoal filter			
Fans, standard conditions	2	8,000 cfm	Optional
Fan pressure	-	5.0-in. H_20	Optional
Fan motors	2	10 hp	Optional
Filters and charcoal filters; roughing, HEPA	2	8,000 cfm	Optional
Steam heating			
Heaters, 25 psig steam	2	400,000 Btu/hr each	Optional

Notes:
1. This value reflects the increase in air side flow rate due to removal of the original plant HEPA Filters.
2. Depends on time of year and containment atmospheric temperature.
3. Based on minimum assumed performance at 271°F containment temp 95°F Service Water temp, 1600 gpm Service Water flow and 64,500 cfm air flow rate.
4. The two exhaust fans are used interchangeably or as backup for:
    1. Ventilation of primary auxiliary building.
    2. Containment building purge system.
5. Purge (25,000 cfm) and primary auxiliary building exhaust (55,500 cfm) are fed into a common plenum.
6. Purge supply (23,000 cfm) and purge exhaust (25,000 cfm) are the nominal, as built, flow rates for the purge system (± 10%).
7. At 72,500 cfm flow rate.

5.3 FIGURES

Figure No.	Title
Figure 5.3-1	Containment Cooling and Ventilation System [Replaced with Plant Drawing 9321-4022]